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SYNBURN--
A FAST-REACTOR FUEL-CYCLE PROGRAM

by

P. A. Pizzica and D. A. Meneley

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ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

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January 1976

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TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

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ABSTRACT

The SYNBURN computer program for fast reactors will calculate all the neutronics necessary to completely characterize the equilibrium cycle as well as the startup to equilibrium cycles. The program's run time is very short and this makes the program suitable for survey of parametric studies. It can search on the cycle time for a specified burnup, for the shim control necessary for criticality as well as feed enrichments and the enrichment ratio among core zones. SYNBURN synthesizes in a very simple fashion the one-dimensional fluxes in radial and axial geometry to achieve an approximate two-dimensional solution which agrees very well with the exact two-dimensional solution when measuring regional integrated quantities.

I. INTRODUCTION

The SYNBURN code was written to provide rapid, approximate two-dimensional solutions in r-z geometry for fuel cycle analyses of fast breeder reactors. It can also compute a one-dimensional solution in slab, cylindrical or spherical geometry. A complete equilibrium cycle calculation can be done, and an arbitrary number of cycles can be calculated with any specified loading as in a startup to equilibrium situation.

There is a great potential savings in computer time as compared to an exact two-dimensional calculation as will be shown in Section VI. This model is intended only for survey and parametric studies. Where much depends upon the precision of each detail of the computation, SYNBURN cannot be relied upon. However, the code can calculate reasonably well quantities which represent averages over reactor regions such as core enrichments, breeding ratios or average core and blanket loadings.

The flux solution method is exactly that of the AIM-5[1] and MACH1[2] codes. The burn matrix routines were taken directly from the CYCLE-1[3] code. The basic model is taken from that described in Ref. 4, as developed for the ARC system at ANL.

The storage limitations of the program are as follows. A maximum of 20 regions is allowed in any one-dimensional flux solution. Therefore 400 regions would be allowed in a two-dimensional computation (However, the core cannot be divided axially.). Scatter-reloading can be accounted for explicitly in a one-dimensional problem. A maximum of 10 refueling cycles is allowed as the life of a fuel subassembly. Any problem is limited to 20 different isotopes. Some or all of these 20 isotopes may appear in each region. Twenty-six energy groups are allowed, 15 downscatter groups and 150 mesh points in any one-dimensional flux solution. Full downscatter is allowed for hydrogen.

Real time accounting and constant-power normalization are used in the burn calculation. The at-power interval may be subdivided into 10 sub-steps. Either direct solution or exponential series expansion may be selected for the burn matrix. Region-averaged group fluxes at the beginning of each sub-interval are used in forming matrix elements. The matrix elements are assumed constant over each sub-interval. The structure of the burn matrix is limited to two independent five-member reaction chains plus a single lumped fission product. There is no allowance for region-dependent active isotope cross-sections.

Available search options are as follows. The cycle time may be adjusted so that the discharge burnup of the most burned fuel batch, averaged over specific regions, reaches a prescribed value or the length of the cycle may remain fixed. The system may or may not have an adjustable control concentration. If it does not, core enrichments are computed such that the system is just critical at the end of the cycle (and probably a noncritical system will exist at the beginning of the cycle). If there is shim control this will be adjusted along with fresh fuel enrichments to achieve a system which is just critical at the end of the substep during which the reactivity is at a minimum, usually the end of the cycle. The shim control at all other points in the cycle will be adjusted to achieve criticality. The ratios of enrichments between core zones may also be adjusted to bring the peak power in different core zones into agreement.

II. MULTIGROUP DIFFUSION SOLUTION*

The basis of the static neutronics model is the one-dimensional flux solution of the AIM[1] series as applied to the MACH1[2] program at ANL. For energy group j, this may be written as:

$$D_j \nabla^2 \phi_j - \Sigma_{t,j} \phi_j + S_{s,j} + \frac{1}{K} S_{f,j} = 0 \qquad (1)$$

D_j - diffusion coefficient $= \frac{1}{3} \left\{ \sum_m n^m \sigma^m_{tr,j} \right\}^{-1}$ (cm)

n^m - atom density of isotope m (nuclei/cm^3 $\times$ 10^{24})

$\sigma^m_{tr,j}$ - microscopic transport cross-section (cm^2/10^{24} nuclei)

ϕ_j - one-dimensional neutron flux (n/cm^2-sec)

$\Sigma_{t,j}$ - total removal cross-section (cm^{-1})

$$= D_j B_j^2 + \sum_{k>j} \left\{ \Sigma_{j \to k} \right\} + \Sigma_{c,j} + \Sigma_{f,j} + \Sigma_{p,j} + \Sigma_{\alpha,j} + \Sigma_{n^2n,j} - 2\Sigma_{n^2n,j \to j}$$

B_j^2 - trasverse leakage correction (buckling) (cm^{-2})

$\Sigma_{j \to k}$ - elastic plus inelastic transfer cross-section from group j to group k (cm^{-1})

$$= \sum_m n^m \sigma^m_{j \to k}$$

$\Sigma_{c,j}$ - (n,γ) cross-section $= \sum_m n^m \sigma^m_{c,j}$ (cm^{-1})

$\Sigma_{f,j}$ - (n,f) cross-section $= \sum_m n^m \sigma^m_{f,j}$ (cm^{-1})

$\Sigma_{p,j}$ - (n,p) cross-section $= \sum_m n^m \sigma^m_{p,j}$ (cm^{-1})

$\Sigma_{\alpha,j}$ - (n,α) cross section $= \sum_m n^m \sigma^m_{a,j}$ (cm^{-1})

*Much of the material for the flux solution part of this section was taken from Ref. 2.

$\Sigma_{n_2n,j}$ - (n,2n) cross section = $\sum_m n^m \sigma^m_{n2n,j}$ (cm^{-1})

$\Sigma_{n2n,j\to j}$ - (n,2n) transfer cross-section from group j to group j (cm^{-1})

$S_{s,j}$ - scattering source = $\sum_{k<j} \Sigma_{k\to j}\phi_k + 2\sum_{k<j} \Sigma_{n^2n,k\to j}\phi_k$ (cm^{-1})

K - effective multiplication factor

$S_{f,j}$ - fission source = $\chi_j \cdot \sum_k \nu_k \Sigma_{f,k}\phi_k$ (n/cc-sec)

χ_j - fission neutron emission spectrum

$\nu_k\Sigma_{f,k}$ - total neutrons emitted per fission in group k = $\sum_m n^m \nu_k^m \sigma^m_{f,k}$

The finite-difference approximations to the diffusion term at space point i are:

$$\left.\frac{d\phi_i}{d\xi}\right|_i = \frac{\phi_{i+1} - \phi_{i-1}}{2\Delta\xi} \tag{2}$$

and

$$\nabla^2\phi = \frac{1}{(\Delta\xi)^2}\left\{\phi_{i+1} - 2\phi_i + \phi_{i-1} + \frac{(N-1)\Delta\xi}{2\xi}\left(\phi_{i+1} - \phi_{i-1}\right)\right\}. \tag{3}$$

where N = 0 for slabs, N = 1 for cylinders, and N = 2 for spheres.

In spherical geometry, trapezoidal integration of functions of the form $\xi^2\phi(\xi)$ leads to large errors near the origin. All such integrals in the code are corrected by the first-order error term of the trapezoidal rule.

The reactor is divided into regions in which D_j, B_j^2, all $\Sigma_{x,j}$, ν_j and $\chi_{f,j}$ are constant. All material constants are nonlinear functions of time through coupling with the isotopic change equations.

Downscatter due to hydrogen is treated by the method described by H. H. Hummel.[5] The scattering source into group j from all groups k above j in energy is then calculated from:

$$S^{H}_{s,j} = \Delta E_j \sum_{k<j} \frac{\Sigma_{rem,\,k}\phi_k}{E_k} \qquad (4)$$

where Σ_{rem} is the removal cross section and E_k is the lower energy bound of group k. To preserve neutron balance, the ΔE_j for the lowest energy group is set equal to E_{j-1}.

At interfaces between regions, the usual flux and current continuity conditions are satisfied using the gradient approximation given by Eq. (2). The arrangement of the difference scheme requires that each region contain at least two mesh intervals.

Three boundary conditions are available at each of the inner and outer boundaries. The general expression is:

$$A\phi_j + B\frac{d\phi_j}{d\xi} = 0 \quad . \qquad (5)$$

The options are as follows:

a. $A = 0$, $B = 1$ (zero gradient);

b. $A = 1$, $B = 0$ (zero flux);

c. $A = 1$, $B = \omega_j$ (homogeneous mixed);

where ω_j is the linear extrapolation length; it is negative at inner boundaries and positive at outer boundaries.

The linear extrapolation lengths of option C may either be input by the user or calculated by the code, according to the equations given by E. R. Cohen.[1] These are simple interpolative formulas which have been fitted to most of the available data for black cylinders and spheres. The equations for extrapolation length on external surfaces are correct at zero radius and at infinite radius; however, the midrange dependence is not based on any detailed calculations (see Ref. 1). For practical radii (>10 cm), the calculation value is very near to the asymptote at infinite radius.

With the input conditions, the calculation proceeds from the reactor outer boundary to the inner boundary, starting with the highest energy group. After the sweep through all groups is completed, a new fission source is calculated from the resultant fluxes. This source is then normalized to a total of one fission neutron in the reactor in the direction of calculation; that is, no integration is carried out over the transverse direction in slab or cylindrical geometry. That is,

$$\text{fission-source integral} = \int_0^{\xi_{outer}} \sum_k \nu_k \Sigma_{f,k} \phi_k \, d\xi = 1.0 \quad . \qquad (6)$$

In slab geometry, this source integral has dimensions of $n/(cm^2\text{-sec})$; in cylindrical geometry, its dimensions are $n/(cm^2 sec)$; in spherical geometry, they are n/sec.

For the first two iterations, the new normalized source distribution is used to calculate fluxes in the following iteration. In the third and subsequent iterations, the starting source at a point for the following iteration is obtained by linear extrapolation from the two previous iterations. The equation is:

$$S'_{f,\ell+1} = S_{f,\ell} + \theta_\ell (S_{f,\ell} - S_{f,\ell-1}) \tag{7}$$

$S_{f,\ell}$ = the normalized resultant source from iteration ℓ,

$S_{f,\ell+1}$ = the source guess for iteration $\ell+1$,

$$\theta_\ell = \theta_0 \left\{ 1 - \ell^{-0.5(\ell-2)} \right\}$$

and

θ_0 = the source extrapolation factor $0 < \theta_0 < 1$.

$S'_{f,\ell+1}$ will not be extrapolated to a negative value. If a negative value is found, it is replaced by zero.

The flux solution iterations are continued until their fixed limit is exceeded or until

$$\left| \frac{k^\ell - k^{\ell-1}}{k^\ell} \right| < \varepsilon \tag{8}$$

where ε is input by the user.

The one-dimensional options of SYNBURN use these equations directly. A crude synthesis approximation is employed in the two-dimensional rz model which uses a combination of one-dimensional flux solutions in slab and cylindrical geometry.

The r-z reactor model is divided into successive annular rings designated as "channels." A one-dimensional axial solution is obtained in each of these channels using a buckling approximation to the radial leakage. The single radial solution uses a buckling approximation to the axial leakage.

A schematic of the radial (r) and one of the channel (z) calculation models is shown in Fig. 1. Given some guess of the axial buckling $B^2_{r,j}$ for each channel, a radial flux solution is found. These fluxes are used to compute the first guess of the radial bucklings for insertion in the channel solution, by the expression,

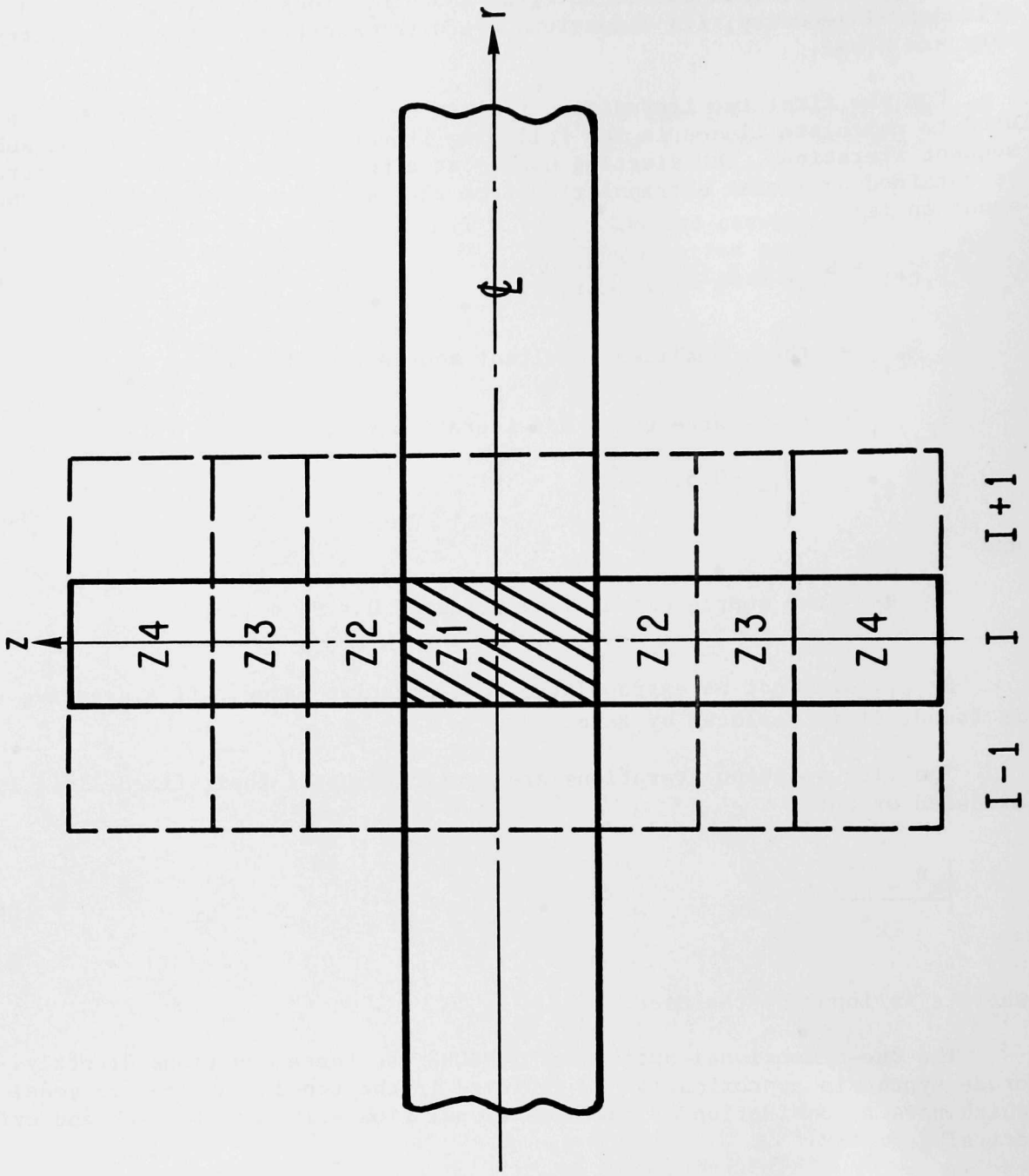

Fig. 1. Schematic of Channel Calculation.

$$B^2_{r,j} = \frac{\int \nabla^2 \phi_j \, dr}{\int \phi_j \, dr} \quad . \tag{9}$$

The integration range is the channel width of channel I.

These radial bucklings are then used in the channel I calculation, but are only appropriate for zone Z1. The main weakness of the model lies, of course, in these transverse leakage approximations, which are known to be at least somewhat inconsistent, (e.g., Ref. 2, Appendix C). Application of the $B^2_{r,j}$ to represent radial leakage in zones Z2, Z3, and Z4 is not usually advisable, particularly (as is often the case) if some of them are negative. The reason is that if one of these zones contains a composition with absorption to leakage balance very different than that of Z1, (in which the $B^2_{r,j}$ were calculated) anomalous results will very likely be obtained from the channel solution. Numerically, this results from very small or negative $\Sigma_{t,j}$ values appearing in Eq. (1) for some j.

In place of the $B^2_{r,j}$, the SYNBURN model usually uses radial buckling values input by the user for all zones other than the Z1 zone although the option exists of using the $B^2_{r,j}$ in any of the Z zones which the user specifies. If zone Z1 is a core zone and zones Z2, Z3 and Z4 are axial blanket and reflector zones, it is not advisable to use the $B^2_{r,j}$ from Z1 for the reasons already stated. However, more reasonable results may very well be produced by using the Z1 $B^2_{r,j}$ in zone Z2 if Z1 is the part of the radial blanket adjacent to the core and Z2 is the upper radial blanket adjacent to axial blanket zones since the absorption to leakage balance is similar (if different) and user-specified $B^2_{r,j}$ would, in many cases, be a worse approximation to the transverse leakage. It follows from the preceeding that the $B^2_{r,j}$ in the axial reflector zones must also be treated separately.

The next iterate of the radial solution uses axial bucklings $B^2_{z,j}$ obtained from Eq. (9), with the range extending over zone Z1. This iterative process continues until either other parameters such as cycle time and core enrichment have reached their desired values or until an upper bound on the number of iterations, fixed by the user, is reached. The radial-axial iterative procedure actually used is complicated by the coupling to the isotopic change equations, the existence of fuel management, a possibly varying cycle time and core enrichment and shim control adjustments. In addition, these solutions must be obtained at several time points in the cycle.

Flux normalization is obtained by normalizing to the same average power in zone Z1 in both radial and channel calculations. This results in a mismatch of average fluxes in some groups. Absolute normalization of power is obtained by integration over all channels.

In order to obtain an expression for the total reactor power in a synthesis problem, given both a radial and an axial flux distribution, the power distribution, P(r,z), is assumed to be a separable function of r and z:

$$P(r,z) = A \cdot Q(r) \cdot T(z) \tag{10}$$

where A is some normalization factor. Thus the total power P^T, is:

$$P^T = \frac{\int_z \int_r P(r,z)\, dr\, dz}{3.1 \times 10^{10} \text{ fissions/watt-second}}$$

$$= \frac{A \cdot \sum_R \left\{ \int_z T_R(z)\, dz \int_{r\epsilon R} Q_R(r)\, dr \right\} \cdot C}{3.1 \times 10^{10} \text{ fissions/watt-second}} \tag{11}$$

where A is the flux normalization factor to be solved for, C is 2π times the core height, and P^T is the total power in watts. The summation is carried out over all radial regions R because $T_R(z)$ and $Q_R(r)$ are variant with R. Thus for each R:

$$\int_{r\epsilon R} Q_R(r)\, dr = \sum_j \left\{ \Sigma^R_{f,j} \int_{r\epsilon R} r\phi_j(r)\, dr \right\} \tag{12}$$

and

$$\int_z T_R(z)\, dz = \frac{\sum_Z \left\{ \sum_j \left[\Sigma^Z_{f,j} \int_{z\epsilon Z} \phi_j(z)\, dz \right] \right\}}{\sum_j \left[\Sigma^{Z_0}_{f,j} \int_{z\epsilon Z_0} \phi_j(z)\, dz \right]} \tag{13}$$

where the summation in the numerator of the expression for T_R is over every axial region Z in the channel corresponding to the radial region R, $\phi_j(r)$ is integrated over region Z, and $\Sigma^R_{f,j}$ is the macroscopic fission cross section for region R and group j; the Z_0 index means the axial region which is included in the radial problem and thus may mean the lower radial blanket in some channels.

In any particular radial flux normalization, the $\int_z T_R(z)\, dz$ for each radial region R and for each time point are either as input (on the first radial pass) or are calculated during the slab problems for the subsequent radial problems on the next radial pass.

In order to normalize fluxes in the slab calculations in a synthesis problem, the normalized average fission density, $\overline{FD}$, is taken from the radial problem for the core or the radial blanket, as the case may be, and for the proper time point, and then the average fission density (unnormalized) obtained from the slab problem for the core or radial blanket is assumed to be the same:

$$\overline{FD} = A \cdot \sum_j \left\{ \Sigma_{f,j}^{Z_0} \cdot \bar{\phi}_j^{Z_0} \right\} \tag{14}$$

where Z_0 refers to either a core or a radial blanket zone, A is the flux normalization factor for which we are solving, and $\bar{\phi}_j$ is the axial region-averaged flux in Z_0 for group j.

In order to normalize fluxes in a one-dimensional problem, Eq. (11) is again solved to obtain the flux normalization factor A, only now all the $\int_z T_R(z)\, dz = 1$. Let R refer to any slab, cylinder, or sphere regions, and, of course, $\int_{r\epsilon R} Q_R(r)\, dr$ and C will take on new definitions according to geometry. In slab geometry,

$$C \cdot \int_{r\epsilon R} Q_R(r)\, dr = \sum_j \Sigma_{f,j} \int \phi_j(r)\, dr \cdot N\pi R^2_{PHYSICAL} \tag{15}$$

where N = 1 for nonreflective slabs and N = 2 for reflective slabs; and in cylindrical geometry,

$$C \cdot \int_{r\epsilon R} Q_R(r)\, dr = \sum_j \Sigma^R_{f,j} \int r\phi_j(r)\, dr \cdot 2\pi H_{core} \tag{16}$$

as before, and for spheres,

$$C \cdot \int_{r\epsilon R} Q_R(r)\, dr = \sum \Sigma^R_{f,j} \int r^2\, \phi_j(r)\, dr \cdot 4\pi \tag{17}$$

As can be seen, this synthesis method must be used very carefully, and its value depends to some extent upon the existence of a more correct model against which to check the results. Such a model is REBUS,[4,6] developed as part of the ARC system at ANL to compute fuel cycle parameters using a true two-dimensional flux solution. Results of a comparison between REBUS and SYNBURN are given in Section VI.

III. SOLUTION OF THE ISOTOPIC CHANGE EQUATIONS*

The fluxes from the diffusion-theory calculation are used to obtain regional one-group effective cross sections for use in the isotopic change

*The material in this section was taken mainly from Ref. 3.

part of the calculation. The spatial group-dependent fluxes are first averaged to obtain regional average fluxes $\phi(j,R)$ for group j and region R. For cylinders,

$$\phi(j,R) = \frac{2\int_{r_{R-1}}^{r_R} r\Phi(j,r)\,dr}{r_R^2 - r_{R-1}^2}, \tag{18}$$

and for slabs,

$$\phi(j,R) = \frac{\int_{r_{R-1}}^{r_R} \Phi(j,r)\,dr}{r_R - r_{R-1}}, \tag{19}$$

and for spheres,

$$\phi(j,R) = \frac{3\int_{r_{R-1}}^{r_R} r^2\Phi(j,r)\,dr}{r_R^3 - r_{R-1}^3} \tag{20}$$

where r_R is the outer radius for region R, and where $\Phi(j,r)$ is the point-dependent flux for group j in region R.

The integrals are evaluated by trapezoidal integration. For example,

$$\int_{r_{R-1}}^{r_R} r\phi(j,r)\,dr = \Delta_R\left\{\frac{1}{2}\left[r_{R-1}\Phi(j,r_{R-1}) + r_R\Phi(j,r_R)\right] + \sum_{r=r_{R-1}+\Delta_R}^{r_R-\Delta_R} r\Phi(j,r)\right. \tag{21}$$

where Δ_R is the spatial interval width for region R given simply by $r_R - r_{R-1}$ divided by the number of spatial intervals in region R.

The $\phi(j,R)$ are used to collapse the microscopic capture and fission cross sections σ_c^i and $\sigma_c^i(j)$ for isotope i and group j. These are given by

$$\langle\sigma_c^i\rangle_R = \frac{\sum_{j=1}^{J} \sigma_c^i(j)\phi(j,R)}{\sum_{j=1}^{J} \phi(j,R)} \quad \frac{\sum_{j=1}^{J} \sigma_c^i(j)\phi(j,R)}{\langle\phi\rangle_R} \tag{22}$$

and

$$\langle \sigma_f^i \rangle_R = \frac{\sum_{j=1}^{J} \sigma_f^i(j)\phi(j,R)}{\langle \phi \rangle_R} \tag{23}$$

where J denotes the number of energy groups and $\langle \phi \rangle_R$ is the effective one-group flux for region R. The bracket is used to denote regional effective one-group values.

The isotopic change calculation is based on the regional effective one-group cross sections $\langle \sigma_c^i \rangle_R$ and $\langle \sigma_c^i \rangle_R$. In the following, to simplify notation, the brackets and regional subscript will be eliminated. Twelve isotopes are considered in the long-term isotopic change calculation. These are ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, fission product pairs, and a burnable poison, numbered respectively 1 through 12. The isotopic change equations can be written in matrix form as:

$$\frac{d}{d\tau}\begin{bmatrix} n^1 \\ n^2 \\ n^3 \\ n^4 \\ n^5 \\ n^6 \\ n^7 \\ n^8 \\ n^9 \\ n^{10} \\ n^{11} \\ n^{12} \end{bmatrix} = \begin{bmatrix} -A^1 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & n^1 \\ C^1 & -A^2 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & n^2 \\ 0 & C^2 & -A^3 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & n^3 \\ 0 & 0 & C^3 & -A^4 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & n^4 \\ 0 & 0 & 0 & C^4 & -A^5 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & n^5 \\ 0 & 0 & 0 & 0 & 0 & -A^6 & 0 & 0 & 0 & 0 & 0 & 0 & n^6 \\ 0 & 0 & 0 & 0 & 0 & C^6 & -A^7 & 0 & 0 & 0 & 0 & 0 & n^7 \\ 0 & 0 & 0 & 0 & 0 & 0 & C^7 & -A^8 & 0 & 0 & 0 & 0 & n^8 \\ 0 & 0 & 0 & 0 & 0 & 0 & 0 & C^8 & -A^9 & 0 & 0 & 0 & n^9 \\ 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & C^9 & -A^{10} & 0 & 0 & n^{10} \\ F^1 & F^2 & F^3 & F^4 & F^5 & F^6 & F^7 & F^8 & F^9 & F^{10} & 0 & 0 & n^{11} \\ 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & -A^{12} & n^{12} \end{bmatrix} \tag{24}$$

where n^i is the atomic density for isotope i and

$$\tau = \int \tau\, \phi(\lambda)\, d\lambda \tag{25}$$

is the integrated flux-time. Also, $A^i = \sigma_c^i + \sigma_f^i$, $C^i = \sigma_c^i$, and $F^i = \sigma_f^i$. ^{241}Pu β - decay is included. We also note that the U-Pu chain is broken at ^{242}Pu, and the Th-U chain at ^{236}U. Also, fission products are not lost; a fission product which captures a neutron is assumed to lead to a neutronically equivalent fission product.

One may write the previous Eq. (24) symbolically as

$$\frac{d}{d\tau}(N) = (M) \cdot (N) \quad . \tag{26}$$

The final and initial atomic densities, N_f and N_i, are related by the matrix (B_f),

$$(N_f) = (B_f) \cdot (N_i) \quad , \tag{27}$$

where the elements of (B_f) are obtained from the analytic solution of the coupled differential equations (24). The form of (B_f) is given by:

$$(B) = \begin{bmatrix}
B_{11} & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\
B_{21} & B_{22} & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\
B_{31} & B_{32} & B_{33} & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\
B_{41} & B_{42} & B_{43} & B_{44} & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\
B_{51} & B_{52} & B_{53} & B_{54} & B_{55} & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\
0 & 0 & 0 & 0 & 0 & B_{66} & 0 & 0 & 0 & 0 & 0 & 0 \\
0 & 0 & 0 & 0 & 0 & B_{76} & B_{77} & 0 & 0 & 0 & 0 & 0 \\
0 & 0 & 0 & 0 & 0 & B_{86} & B_{87} & B_{88} & 0 & 0 & 0 & 0 \\
0 & 0 & 0 & 0 & 0 & B_{96} & B_{97} & B_{98} & B_{99} & 0 & 0 & 0 \\
0 & 0 & 0 & 0 & 0 & B_{10,6} & B_{10,7} & B_{10,8} & B_{10,9} & B_{10,10} & 0 & 0 \\
B_{11,1} & B_{11,2} & B_{11,3} & B_{11,4} & B_{11,5} & B_{11,6} & B_{11,7} & B_{11,8} & B_{11,9} & B_{11,10} & 1.0 & 0 \\
0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 & B_{12,12}
\end{bmatrix}$$

A few of the elements of (B) are given below to illustrate the structure of the solution.

$$B_{11} = E^1$$

$$B_{21} = \frac{C^1}{A^{21}} E^{21}$$

$$B_{22} = E^2$$

$$B_{31} = \frac{C^1C^2}{A^{21}A^{31}} E^{31} + \frac{C^1C^2}{A^{12}A^{32}} E^{32} \qquad (29)$$

$$B_{32} = \frac{C^2}{A^{32}} E^{32}$$

$$B_{33} = E^3$$

where

$$E^i = \exp(-A^i \cdot \tau)$$

$$E^{ji} = \exp(A^i \cdot \tau) - \exp(-A^j \cdot \tau)$$

$$A^{ij} = A^i - A^j$$

It is clear from Eq. (29) that there exists the possibility of accidental cancellation of cross sections with consequent indeterminacy of the elements of (B). Another possibility is that inaccuracies may arise due to round-off errors.

An alternate series solution to Eqs. (24) or (26) is possible which avoids these problems. Symbolically, this alternate solution can be written:

$$(N_f) = \left\{\exp[\tau \cdot (M)]\right\} \cdot (N_i) = \left\{(1) + \tau \cdot (M) + \frac{\tau^2}{2!} \cdot (M)^2 + ..\right\} (N_i) \qquad (30)$$

The series in the curly bracket is equivalent to the matrix (B_f). The series method may be bothersome in the case where many terms are required for convergence, as for example in systems requiring large values for τ or having large values for some of the elements of (M).

If one considers the average over τ of the atomic densities, then the average values:

$$(N_a) = \frac{1}{\tau} \int_0^\tau (N)\, d\lambda \quad , \qquad (31)$$

will be related to the initial values by the matrix (B_a),

$$(N_a) = (B_a)(N_i) \quad , \qquad (32)$$

where, as in Eq. (27) the elements of (B_a) are due to the analytic solution for the average atomic densities. Again, alternatively the series solution can be used and is given by:

$$(N_a) = \left[\tau \cdot (M)\right]^{-1} \cdot \left\{\exp\left[\tau \cdot (M)\right] - (1)\right\} \cdot (N_i)$$

$$= \left\{(1) + \frac{\tau}{2!} \cdot (M) + \frac{\tau^2}{3!} \cdot (M)^2 + \ldots\right\} \cdot (N_i) \qquad (33)$$

Again, (B_a) is given by the series in the curly bracket.

By means of an input option, the user may designate that either the analytic method or the series method be used. In the latter case, two possibilities exist in the event that more terms in the series are required than are allowed by the program. In one case, the analytic method will be used for those regions in which the series limit is exceeded. In the other case, if the series limit is exceeded for any region during running of the problem, the analytic method is used for the remainder of the problem. The "final" or "average" solutions will be obtained automatically by the code, depending upon the part of the program which is involved.

The series expressions for (B_f) and (B_a) given in the curly brackets of Eqs. (30) and (33) respectively are evaluated out to and including the matrix:

$$(\text{ANEXT}) = \frac{\tau^2}{8!} (M)^8 \qquad (34)$$

for the case of (B_f) and out to and including the matrix

$$(\text{ANEXT}) = \frac{\tau^7}{8!} (M)^7 \qquad (35)$$

for the case of (B_a). The code then evaluates,

$$\text{BERROR} = \sum_{m,n} \left|(\text{ANEXT})_{mn}\right| \quad . \qquad (36)$$

If BERROR $\leq$ BEPS, the solution is converged, and the elements of the matrix (B) are set equal to the elements of the matrix evaluated by the series. If BERROR > BEPS, the next matrix in the series is evaluated, and the test is repeated. If convergence is not obtained when the series contains the matrix

$$\frac{\tau^{15}}{15!} (M)^{15}$$

in the case of (B_f), or the matrix

$$\frac{\tau^{14}}{15!}(M)^{14} \quad ,$$

for the case of (B_a), then, depending on the option used, either the analytic method is used for the remainder of the problem, or it is used for the region where the series solution failed, and then the series solution is tried on the region, or the series will continue until convergence.

IV. PROCEDURE FOR SOLVING THE EQUILIBRIUM PROBLEM

It will be helpful first to define certain terms as an aid in describing regional and subassembly compositions as a function of time. The problem is to develop an indexing system which will permit location of each discrete fuel element in space over the whole period during which it is in the reactor. It may be loaded into the core at a particular position and irradiated for some fixed cycle time in a flux which is determined by the interaction of all the fuel elements in the core, as well as by control shifts. After this initial "burn," the element may or may not be repositioned in space. In any case, if any of the fuel in the core is moved, charged, or discharged, the irradiation rate will undergo a discontinuous change at the so-called "shuffling step." After several burn-shuffle sequences, the element is discharged from the system.

The spatial movement of fuel elements is constrained by requirements of volume preservation. Whenever elements are shuffled (repositioned) in space, other elements of equal volume must be removed from their position and inserted in other locations, or discharged, and others must be moved to occupy the space vacated by the original fuel elements. This linked series of motions is the basis of the indexing system.

A first index for each of these fuel elements is given the name "material." A material consists of all the fuel elements in the reactor at a particular time, which are linked by a single shuffling sequence or "path." The individual fuel elements, the "material-stages," are identified by the second index. This second index is 1 for a fuel element which has just been charged into the core, 2 for an element which has been shuffled once, and so on up to the maximum stage index, after which the fuel element is discharged. A third index, the "region," is used to locate each material-stage in space. A region is the volume over which the average fluxes are computed in obtaining the fuel burnup. A region in a cylindrical model may be an annular ring containing a number of fuel elements of different composition and burnup history. Thus several different material-stages may be assigned to a single region, with the constraint that the sum of the volumes of all material-stages in a region equals the volume of the region.

A particular application of the method is illustrated in Fig. 2, a plan view of a 37-subassembly core. The subassemblies marked "A" are to be charged into ring 4, shuffled to ring 3, then 2, and discharged. The "B" subassemblies are to be charged into ring 3, moved to 4, and then to ring 1, at every other burn step. Similar patterns apply to materials "C" and "D". Table I shows

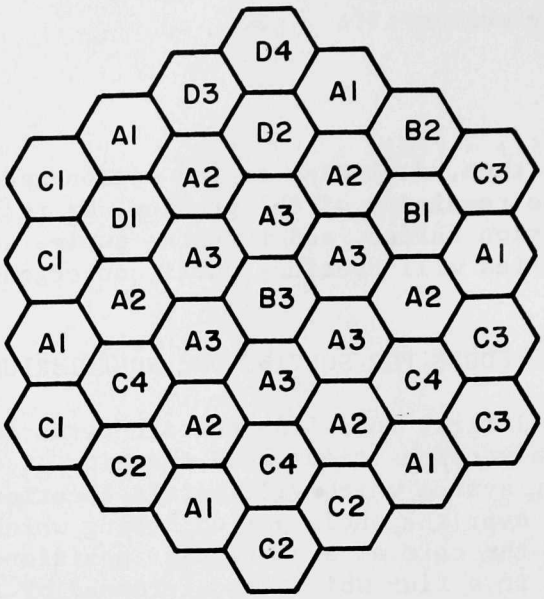

Fig. 2. Example of Loading and Fuel Shuffling Pattern

TABLE I. Example of Indexing System Defining In-core Fuel Motions

MATERIAL	SUBASSEMBLIES PER STAGE	STAGE 1 2 3 4 REGION
A	6	4 3 2
B	1	3 4 1
C	3	4 4 4 3
D	1	3 3 4 4

the input indexing system for this core, assuming that it is to be computed as a four-region annular system in one-dimensional cylindrical geometry. Material-stage and region volumes are defined separately.

In the case of equilibrium problems, this indexing system is extremely useful. Since by definition the fueling and shuffling are held constant, the material-stages present in the core at any time can be considered as different phases of the burnup history of a single fuel lump. This greatly simplifies the equilibrium cycle computation for scatter-refueled cores. It is obvious that material shuffling cannot occur in a two-dimensional problem since SYNBURN cannot compute all fluxes in all regions axially and radially simultaneously. Full allowance is made, however, for material shuffling in one-dimensional problems.

In order to form regional atom densities, the region residence data for the stages of each material must be defined, as well as information about the poison and diluent in the core. Region-averaged compositions are formed as follows:

$$N_i^R = \sum_{j \epsilon R} (N_i^j \cdot V^j/V^R) + N_{P,i}^R \cdot x + N_{D,i}^R \cdot (1 - x) \qquad (37)$$

where N_i^R is the region-averaged atom density for region R and isotope i, N_i^j is the atom density for the material-stage j, V^j is the volume of material-stage j, and V^R is the volume of region R. What is meant by the summation of all j belonging to R is that only those material-stages j which reside in R will contribute their N_i^j to N_i^R. $N_{P,i}^R$ and $N_{D,i}^R$ are the poison and diluent atom densities, respectively, input by the user for region R and isotope i (all those regional isotopes not specified by the user are set to zero by the code), and x is the factor which is determined by the code to compute the precise poison and diluent concentrations at each time point.

The condition of equilibrium will first be described and then it will be explained how the code attempts to approximate this condition. When the reactor attains equilibrium, successive cycles are characterized by identical reactor behavior. A cycle is, of course, the time between two successive shutdowns. If a critical system is to be maintained at all times, an enrichment of the charge fraction and a minimum control concentration consistent with this enrichment are necessary. The equilibrium cycle time must be such that the required burnup in the discharge fraction is obtained given a fixed reactor power level.

It might be best at this point to consider an example. Suppose there is an equilibrium cycle time of 200 days and a core with one enrichment zone which is refueled in thirds and one radial blanket zone which is refueled in sixths. Thus, at the start of a cycle, one-third of the core is unburned, one-third is one-third burned, and one-third is two-thirds burned. And at the end of the cycle, the residence time for the third fraction will be 600 days. Similarly, in the blanket, at the beginning of life, one-sixth is fresh, one-sixth is one-sixth burned, and so on, and the last sixth is five-sixths burned. The last stage will have resided in the blanket for 1200 days when it is discharged at the end of cycle. Equilibrium is defined in this system as that

set of reactor conditions which produces identical compositions, fluxes, etc. for the same point in all successive cycles. This means that the fresh feed fraction will be the same at each shutdown and also that the discharge fraction in each region at each shutdown is identical with the discharge fraction from the previous shutdown. What is implied in the previous sentence is that the enrichment (control) and the flux variation over the cycle are identical to the variation over the previous cycle.

The following is a brief description of each step in the solution of a synthesis problem, including an explanation of the techniques employed, the simplifying assumptions made, and the parameters which are held fixed. It should be obvious from this description how a one-dimensional solution proceeds.

Besides data about the geometry, the user specifies the total reactor power level, either a fixed transverse leakage or a starting guess for it, either a fixed burn time or a required average burnup, an initial guess or a fixed value for the relative power production in the axial blanket and upper radial blanket zones, relative amounts of control and diluent in the core, inactive isotope concentrations and the number of stages for each material, and, in order to calculate the active isotopes, the fuel volume fraction and the isotopic distribution are given for each material as well as the ratio of enrichments between core zones in a multi-zoned core, unless this is to be determined by the power-shape optimization option.

Since the user only provides information for the stage one atom densities, these densities are loaded into all stages, and region-homogenized atom densities are formed on this basis with the full amount of diluent and no poison in the system. The flux distribution computed for these compositions is used to burn the stage one densities for a time equal to N - 1 multiplied by the burn time (as input) to form stage N. With the new compositions, a new flux distribution is computed. These fluxes are used to burn the system to the end of the input cycle time, where a new k-effective is computed. If the difference between the two k-effectives indicates a decrease, the difference is the amount above one to which an uncontrolled enrichment search is made. If the k-effective increases, the search is made to a k-effective of one. The user also has the option of specifying an unpoisoned beginning-of-cycle k-effective to which to search. At this point the cycle time is adjusted, based on the average discharge burnup obtained from the above approximation to the end-of-cycle compositions. The user specifies a certain discharge fractional burnup averaged over a number of specified materials. Burnup is defined as

$$\frac{\sum_{i \epsilon M} FP_i \cdot V_i}{\sum_{i \epsilon M} HA_i \cdot V_i}$$

where i is the material within the group of specified materials M, FP_i is the fission product pairs concentration in the discharge fraction of material i, HA_i is the total charge fuel concentration for material i, and V_i is the subassembly volume for material i.

Beginning-of-cycle compositions are again formed, and a search is done with no control in the system on the enrichment in the first stages to the desired k-effective. There is more involved, however, than merely altering the enrichment of the first stage, because when the first stage is altered, the higher stages must obviously be changed as well. This, of course, requires some fluxes at which to burn the first stage. During the enrichment search, these fluxes are taken from the previous pass to form new higher-stage compositions for the new stage one enrichments after starting the search with some arbitrary fluxes (actually these are the end-of-cycle fluxes generated above). There is yet another factor involved. One ought ideally to take into account the effects of the alteration of the higher-stage compositions and their effect on the flux distribution. In other words, every time the enrichment of the first stage is altered, the higher stages formed are not the correct compositions, since the fluxes used to generate them were computed on the basis of the old stage one enrichment. Therefore, what one ideally ought to do is to iterate a few times on the beginning-of-cycle fluxes every time the enrichment is altered. That is, the higher-stage densities should be computed as above and then new region-homogenized atom densities and thus new fluxes should be formed with which to form new higher-stage densities, and so on for an arbitrary number of times. But this technique would be useless because there are other factors which limit the accuracy of predicting these beginning-of-cycle higher-stage compositions. First, in the first enrichment search, there is no means of introducing the time variance of the flux, as there is in subsequent searches (this will be discussed later) and second, the control distribution will only be approximate because of its effects on the higher stages. However, this iterative process as described above is, to a certain extent, included in the actual enrichment search, if one can assume that the convergence criterion on the search is reasonably tight.

At the completion of the search, the system is supercritical (unless the search was to a k-effective of one) with no control, and now a search is done, given the relative amounts of control in the core zone as input, to add poison (and remove diluent, of course, which was in the core in the full amount, as specified by the input during the enrichment search) until a critical system is attained. Now presumably the flux distribution has been altered, and again the effects on the higher-stage compositions must be considered. Therefore, with the fluxes from the converged control search, new higher-stage densities are formed, and with the new composition, a new set of fluxes is calculated. The system is probably not quite critical at this point, but presumably this approximation to the beginning-of-cycle system is not so bad as to require another enrichment search.

If the user so specifies it, after the enrichment search at the beginning-of-cycle at each radial pass, the ratio of core enrichments (in a multi-zoned core) is altered to match peak power in each zone.

The last flux distribution computed above is used to burn the system over the first time step. At this point, the control in the core is adjusted to restore criticality. The fluxes from the converged control search are then used to burn the system over the next time step. Control is then adjusted again at the end of the step to again restore criticality. The process is repeated until the end of the cycle, where the final control search is done. If, at the end of any step in the cycle, a critical system cannot be obtained even after all the poison available has been removed, the fluxes from the

off-critical system are used to burn over the next step, and fluxes are computed at all subsequent time points with no control in the system and are used to burn the system over the next time step. In a synthesis problem, group- and region-dependent bucklings in the radial direction, as well as the normalized average power for each region, are calculated at each time point for use in the slab problems following.

A word of explanation is required here concerning the representation of poison and diluent in the reactor. Two things are fixed by the user in the input. These are the relative distribution of the control among the regions of the core and also the amount of diluent relative to the poison. That is, the user inputs atom densities for both the poison and diluent, and these densities correspond to the same volume fraction. Therefore, if the atom density of poison in region R is represented by $x \cdot N^R_{P,i}$, where $N^R_{P,i}$ is the user's input concentration of poison for isotope i and region R and x is some factor determined by the code (which is not region-dependent, but is time-dependent as control varies over the cycle), then the diluent atom density is represented by $(1-x) \cdot N^R_{D,i}$, where $N^R_{D,i}$ is the user's input atom density for the isotope i of the diluent in region R. It is clear that ideally the code should determine a value for x which is at a maximum of one and at a minimum of zero over the cycle. The user, however, will probably find it difficult to even come close to such values for N_D and N_P. Thus, if the input Np is, say, twice as great as the maximum determined to be necessary by the code (its minimum is, of course, zero), then there will be N_D as a maximum diluent concentration and $.5 \cdot N_D$ as a minimum. And if N_P is half the amount required, then the poison varies from $2 \cdot N_P$ to zero, and the diluent varies from $-N_D$ to N_D. The negative sign can have one of two effects: it can subtract the N_D concentration from the constant concentration input in the inactive isotopes, or it can show up as a negative concentration if there is no constant density for the particular isotope. The negative value for N_D can be eliminated in two ways. One is by guessing a large control volume fraction to begin with, if some reasonable upper bound can be determined, or the user can elect to specify only N_P and N_D will be set to zero by the code.

It should be noted that during this first radial problem, the transverse bucklings for each group, region, and time point, and the power increment factors for each region and time point used are, of course, those input by the user. Therefore, the results from this first set of problems cannot be any better than these input parameters.

At the end of the cycle, the beginning-of-cycle control volume fraction is adjusted according to what the value is at end-of-cycle if there is any control left in the system. That is, the beginning-of-cycle volume fraction is adjusted such that there is no poison (actually the code requires for convergence that the minimum amount of poison in the system lie between zero and $2 \cdot \varepsilon \cdot N_P$, where ε is the convergence criterion input by the user) at end-of-cycle. The preceding applies if the unpoisoned system k-effective is consistently decreasing over the cycle. However, if it is determined that the unpoisoned k-effective is increasing over all or part of the cycle, the time point at which the unpoisoned k-effective (or the control volume fraction) is at a minimum is determined, and the beginning-of-cycle volume fraction is adjusted according to the amount of the difference. If, during any search over the cycle, there is not enough control available so that its removal will

restore criticality, then the adjustment of the initial control volume fraction is determined by the difference between the system k-effective and one. Also at the end-of-cycle, the cycle time is adjusted based upon the average of the discharge burnups in the specified materials.

Next, a number of slab calculations are done, one for each radial region where it is specified to do so. In each slab problem, the transverse bucklings can be supplied from any or all of three sources. The group-, region-, and time-dependent radial bucklings from the previous radial problem are, of course, used in the core and radial blanket regions adjacent to the core in the respective slab problems. And, in an upper radial blanket region, these bucklings are usually from the lower radial blanket zone corresponding to it. In the axial blanket zones, the user will want to specify some arbitrary values for the transverse bucklings, and in the axial reflector zones, the blanket transverse bucklings may be used, or the leakage may be set to zero. In each slab calculation, the fluxes are normalized by equating the normalized average power in the zone which was included in the radial problem to the unnormalized average power calculated in the slab problem.

The main problem in calculating the isotopic change over the cycle for these slab problems is obtaining a beginning-of-cycle composition. The first stage in core zones where an enrichment search was done is obtained from the result of the enrichment search in radial geometry and the first stages in the core regions where no enrichment search was done, and in the radial blanket zones, the stage one is as input. The higher stages in the core and radial blanket zones could conceivably be taken from the radial problem, but due to procedural difficulties, this is not done. Also, to obtain the higher stages in all the slab regions, one could conceivably (after the first set of slab problems) save the time-dependent average fluxes from the previous set of slab problems and use these to burn the stage ones; but, again due to procedural difficulties, this is not done. Instead, another method was used. The results from this method were entirely satisfactory and seemed to bear out the presumption that there was little to be gained by surmounting the procedural difficulties referred to above. In every slab calculation, in order to obtain the beginning-of-cycle composition, every region is first loaded with the stage one densities, and the flux distribution produced is used to burn these first stages to form the higher stage. Next, the new region compositions obtained with the new higher stages are used to form new fluxes, which are used to form new higher stages, and so on. Altogether there are four repetitions of this procedure; i.e., a flux computation, together with a recalculation of the higher stages, is done four times in the beginning of each slab problem. A fifth flux distribution is then computed, and this is used to burn the system over the first time step. At the end of the time step, new fluxes are calculated and used in turn to burn over the subsequent time step; and this is repeated until the end-of-cycle, where the last flux calculation is done.

In those slab problems where it is applicable, the control density (which is a function of time) is taken from the radial problem, and in axial blanket zones above those core zones which have control in them, the maximum amount of control which the corresponding core zone contained during the cycle may be placed into these zones at the user's option. With respect to the radial problems, the significance of the slab problems is that better approximations are made for the axial bucklings which are group-, region-, and time-dependent

and for the fraction of total power generated in the axial blanket and upper radial blanket as a function of time. It should be noted here that, in radial regions where no slab problems are done, the transverse bucklings and the power increment factors are those input by the user. Procedurally, all subsequent slab calculations are identical to the first. The user specifies a maximum number for the number of sets of slab problems done, and if this number is zero, none will be done until the edit pass.

After these slab calculations are completed, the second set of radial problems is begun. The second and subsequent radial passes are identical procedurally with the first (of course, it uses the new axial bucklings and power distributions computed in the slab problems), except in the manner in which the higher-stage densities are computed. To start the second radial burn, the resultant enrichment from the enrichment search on the previous radial pass is used to compute the stage one compositions. Now the time-dependent average fluxes computed during the previous radial pass are used to burn these stage one densities to form the higher stages. Region-homogenized compositions are thus formed, and a flux distribution is calculated. The enrichment of the first stage is here altered on the basis of the system k-effective obtained. At this point in the enrichment search, after this first flux calculation, the method of computing the higher stages is changed. Instead of directly using the fluxes calculated on the previous radial pass, now the beginning-of-cycle fluxes computed above are used along with the previous fluxes to compute a new set of time-dependent fluxes. To be more precise, exactly what is obtained from the previous radial pass is one-group region-averaged fluxes and one-group collapsed regional cross sections. These values are, of course, time-dependent. Now the cross sections are used precisely as they are on all subsequent enrichment search passes, but the fluxes are updated as follows: it is assumed that the ratio between the new time-dependent fluxes will be the same as the ratio of the ones from the last pass, and therefore we take the initial one-group fluxes computed above and update them with the ratios of the one-group fluxes from the last pass. For example, since we already have the beginning-of-cycle fluxes, we form the new one-group fluxes for each region at the second time point by multiplying these first time-point fluxes by the ratio of the old time-point two fluxes for the region to the old time-point one fluxes for the region. Thus, although we are using the same collapsed cross sections, we have created an updated set of one-group fluxes for each region. These updated fluxes and the old cross sections are used to form the higher-stage densities, and a new k-effective is computed. Again, the stage one densities are altered, and now the new higher stages are formed by updating the new fluxes, and the process continues to convergence.

The code defines convergence as a direct result of two factors: a stable burn time which produces the required burnup (if one is specified) and a core feed enrichment which produces the minimum amount of control necessary. Convergence is indirectly a result of three other factors, but the reason that this distinction is made here is that the solution is defined as complete when these two criteria are satisfied (along with one other criterion, which is that a minimum of two radial passes must be completed). The flux values as a function of time must be included in the convergence process as they will affect the beginning-of-cycle composition. Two other factors involved in convergence are the axial bucklings and the relative power production in the axial blanket and corners. These are recalculated during each set of slab problems for use in the succeeding radial calculation.

In most problems, a set of slab calculations will be done after only the first few radial problems, and not after all radial problems. Therefore, in this case, "convergence" becomes easier to define, since the axial bucklings and power splits provided by the slab problems will be held fixed at an arbitrary point in the solution, and then two of the five variables involved in convergence are no longer changing. Therefore, in such problems, the solution will be determined relatively quickly after these first slab problems are done; the burn time and core charge enrichment, as well as the time-dependent fluxes, will stabilize, and convergence within the limits specified will be achieved. When it is determined, at the end of a particular radial problem, that the above criteria have been satisfied, one more radial problem is done, and after that another set of slab calculations, and during these calculations, a complete summary of reactor behavior is printed out. It should be noted at this point, however, that the parameters printed out in the last set of slab calculations will be slightly different from the parameters that were computed in the second-to-last set of slab calculations. The reason for this, of course, that the radial bucklings and other information used in these new slab problems are not the same as those used by the previous slab calculations.

V. PROCEDURE FOR SOLVING THE STARTUP TO EQUILIBRIUM PROBLEM

The startup calculation is an extension of the equilibrium calculation. The time between shutdowns is fixed and is the same as the final equilibrium cycle time. The startup calculation will handle a variety of problems including, for example, those where the fuel and the isotopic distribution must be altered during startup. A routine exists which will optimize the power distribution by altering the ratios of enrichments between core zones just as in the equilibrium calculation. The computational methods which are applicable to both problems are identical. For example, the calculation and use of time-and-region-varying group dependent bucklings to represent the transverse leakage and the method of normalizing the fluxes are the same in both calculations. The startup calculation is done in two dimensions via the synthesis model and no option exists for doing a one-dimensional problem.

The axial bucklings and the power splits corresponding to equilibrium beginning-of-cycle are used in the first startup cycle calculation. The relative variation of control (not the absolute volume fractions) over the equilibrium cycle is used to compute control volume fractions during each startup cycle.

To obtain the initial loading for startup, the composition in each core zone is graded in N enrichments (where N is the number of stages in that zone) and the isotopic distributions are specified as input. The ratios between these enrichments within a given core zone are the same as the ratios between the stages at the beginning of the equilibrium cycle. The radial blanket is optionally loaded with a fixed composition or with enough ^{235}U to produce the same average power as in the equilibrium beginning-of-cycle for each radial blanket zone. This amount of ^{235}U is evenly distributed among the stages in each zone.

As initial approximations for the relative power produced in the axial blanket and upper radial blanket and for the axial bucklings, the values computed for the equilibrium beginning-of-cycle are used at all time points for the first radial calculation since these equilibrium compositions in the axial blanket and upper radial blanket are the best approximation there is to any point in the first startup cycle.

With this initial loading, a flux calculation is done in radial geometry and the burnup to mid-cycle is computed. With these first approximations to the mid-cycle composition, another flux calculation is done. These fluxes are used to burn the B.O.L. compositions again to mid-cycle. A k-effective calculation is now done at mid-cycle with the mid-cycle control in the system. The B.O.L. enrichments are now adjusted on the basis of this k-effective and the fluxes just calculated are used to burn the new B.O.L. compositions to mid-cycle where a new k-effective is computed. The B.O.L. compositions are again adjusted. The search on enrichment continues until criticality is attained at mid-cycle.

On the basis of the final flux distribution calculated during the enrichment search, the ratios of enrichment between core zones is adjusted to optimize the power peaking. The ratios of enrichments between stages, however, remains fixed. After this another enrichment search is done to obtain a k-effective of one at mid-cycle. It is assumed that the power distribution is not altered significantly after the second enrichment search and no further alteration of enrichment ratios is done here.

The system is now burned over the entire cycle using the last values for the fluxes computed at mid-cycle. The control is varied over the cycle and radial group-dependent bucklings for each region and time point are computed. Fluxes are normalized using the numbers for the relative power generated in the axial blankets and upper radial blanket calculated during the equilibrium cycle.

Next a series of slab calculations is done, one for each radial zone. The initial axial blanket loading is specified as is the upper radial blanket composition. Mid-cycle compositions are taken from the radial problem for those axial zones which were included in the radial calculation. Bucklings are taken from the radial calculation and the average power from the radial problem is used to normalize the fluxes in the slab calculations. These compositions are used to compute a flux distribution which is a first approximation to the mid-cycle conditions. These fluxes are used to burn the B.O.L. compositions to mid-cycle where another flux distribution is computed. Again these fluxes are used to burn the B.O.L. atom densities to mid-cycle. This iterative procedure is carried on three times and the last fluxes computed are used to burn the B.O.L. compositions over the entire cycle. The bucklings and flux normalization factors for each time point are taken from the radial calculation. And, at each time point, bucklings and power splits are computed for the next radial calculation.

In order to utilize these better approximations to transverse leakage values and power splits, another radial calculation is done as above using these newly calculated parameters. After this, another set of axial calculations is done as above only now using the new radial bucklings and flux

normalization computed in the second radial calculation. During these last radial and axial calculations, a summary of reactor characteristics is printed out in the same form as that for the equilibrium cycle. The preceding constitutes the calculation for one cycle of startup.

The reactor is now reloaded with the specified fuel and distribution; and the enrichment of the charge fraction in the core is varied until criticality is achieved. The calculation for the second and subsequent cycles proceeds as does the first with two exceptions: the axial bucklings and power splits for the first radial calculation come from the last slab calculation in the previous cycle; in order to optimize the power peaking, the ratio of enrichments between only the charge fractions is varied, of course.

The code will continue the calculation for any number of cycles. The fuel for the initial loading or the charge fraction may be specified in any manner at all. For example, a reactor may be loaded initially and reloaded with plutonium out to equilibrium or it may be loaded initially with ^{235}U and starting with the second cycle reloaded with plutonium out to equilibrium, or, perhaps it is loaded initially with ^{235}U and reloaded for two more cycles with ^{235}U and then starting with the fourth cycle is reloaded with plutonium and subsequent reloadings are the same. For each reloading as well as for the initial loading the isotopic distribution is specified and thus may be varied for any of these.

VI. RESULTS OF A COMPARISON WITH AN EXACT TWO-DIMENSIONAL SOLUTION

The following tables list the results of a comparison between results obtained from REBUS[6] and SYNBURN, for an equilibrium cycle calculation. Tables II through VIII give results of a comparison between SYNBURN and REBUS, where the problem in REBUS was defined such that it approximated as closely as possible the SYNBURN case, except, of course, for the true two-dimensional flux solution differing from the synthesis method. This was supposed to check the accuracy of the synthesis method while eliminating extraneous factors. Thus three capabilities of REBUS were not used in this comparison. First, region-dependent cross-sections were not used in the REBUS calculation. In SYNBURN, the core cannot be divided axially (although the axial blanket can) and using this feature of REBUS would have changed the results slightly. Thirdly, the isotopic change equations in REBUS are much more comprehensive than the ones in SYNBURN.

REBUS was run with only one burn step because it was thought that this corresponded most closely with a SYNBURN calculation using two time steps, since REBUS uses a flux averaged over the time step and SYNBURN uses only the flux at the beginning of the step to burn over the step.

The REBUS execution took 28 minutes while SYNBURN took 21 seconds. Both had the same geometrical divisions which are illustrated in Fig. 3. Both used a fixed cycle time of 274 days. REBUS had better initial guesses for both control volume fractions and core enrichments. Both had the same mesh spacing and used the same 8-group cross-section set. In both cases 1 control rod was smeared in region 1 and 12 control rods were smeared in region 5.

TABLE II. Comparison of Core Charge Isotopic Masses (in Kilograms)

	REBUS						SYNBURN					
Region	^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
1	0.4	53.2	9.2	2.6	1.4	0.3	0.4	53.0	9.3	2.6	1.4	0.3
2	0.7	106.4	18.3	5.2	2.7	0.7	0.7	106.1	18.5	5.3	2.8	0.7
3	0.7	106.4	18.3	5.2	2.7	0.7	0.7	106.1	18.5	5.3	2.8	0.7
4	3.3	479.0	82.4	23.5	12.4	2.9	3.3	477.3	83.4	23.8	12.5	3.0
5	1.5	212.9	36.6	10.4	5.5	1.3	1.5	212.1	37.1	10.6	5.6	1.3
6	4.9	712.1	199.2	56.8	29.9	7.1	4.9	708.1	201.5	57.5	30.2	7.2
Inner Core	6.6	957.9	164.8	46.9	24.7	5.9	6.6	954.6	166.8	47.6	25.1	6.0
Outer Core	4.9	712.1	199.2	56.8	29.9	7.1	4.9	708.1	201.5	57.5	30.2	7.2

TABLE III. Comparison of Reactor Loading (in Kilograms)

Region	^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	F.P.
			REBUS - BOC				
Inner Core	15.1	2759.3	461.6	155.6	60.1	19.5	146.9
Outer Core	12.3	2080.5	543.2	180.0	77.2	22.8	111.8
Axial Blanket	8.8	4946.2	127.6	6.1	0.2	--	27.2
Radial Blanket	31.2	16052.4	244.1	6.8	0.1	--	41.0
			SYNBURN - BOC				
Inner Core	15.1	2652.7	465.8	156.7	61.1	19.6	145.1
Outer Core	12.3	2070.8	550.1	181.6	78.4	23.0	109.8
Axial Blanket	8.8	4952.2	122.5	5.6	0.2	--	25.5
Radial Blanket	31.0	16044.1	247.4	6.9	0.1	--	42.4
			REBUS - EOC				
Inner Core	11.1	2648.0	431.8	168.0	49.0	20.8	286.1
Outer Core	10.1	2025.4	494.0	188.0	66.5	24.0	217.6
Axial Blanket	7.9	4883.6	172.7	9.8	0.3	--	41.5
Radial Blanket	29.5	15938.0	334.7	11.1	0.2	--	61.8
			SYNBURN - EOC				
Inner Core	11.2	2644.5	434.7	168.8	49.9	21.0	283.2
Outer Core	10.1	2018.1	500.6	189.4	67.8	24.2	213.6
Axial Blanket	8.0	4891.6	166.9	9.0	0.3	--	39.0
Radial Blanket	29.4	15928.0	339.2	11.2	0.2	--	63.8

TABLE IV. Comparison of Power Fractions by Region

Region	BOC		EOC	
	REBUS	SYNBURN	REBUS	SYNBURN
1	0.03433	0.03439	0.02912	0.02858
2	0.06807	0.06796	0.05738	0.05630
3	0.06573	0.06561	0.05660	0.05568
4	0.25511	0.25400	0.24033	0.23886
5	0.08980	0.09001	0.09849	0.09912
6	0.37857	0.37758	0.37519	0.37742
7	0.05834	0.05937	0.07710	0.07841
8	0.00245	0.00211	0.00298	0.00249
9	0.00469	0.00377	0.00575	0.00442
10	0.00451	0.00486	0.00560	0.00574
11	0.01549	0.01226	0.02035	0.01596
12	0.00528	0.00461	0.00733	0.00697
13	0.01164	0.01085	0.01579	0.01462
14	0.00598	0.00772	0.00798	0.00913
Inner Core	0.51304	0.51197	0.48192	0.47854
Outer Core	0.37857	0.37758	0.37519	0.37742
Axial Blanket	0.04406	0.03846	0.05780	0.05020
Radial Blanket	0.06432	0.06709	0.08508	0.08754

TABLE V. Comparison of Regional Conversion Ratios

Region	BOC		MOC[a]	
	REBUS	SYNBURN	REBUS	SYNBURN
1	0.6864	0.6825	0.7139	0.7141
2	0.6863	0.6813	0.7132	0.7123
3	0.6883	0.6823	0.7146	0.7122
4	0.6745	0.6683	0.7002	0.6963
5	0.6533	0.6483	0.6835	0.6784
6	0.4348	0.4307	0.4583	0.4541
7	5.870	5.864	5.006	5.014
8	3.639	3.700	3.104	3.148
9	3.707	3.921	3.162	3.336
10	3.751	3.424	3.201	2.917
11	4.240	4.593	3.615	3.911
12	4.756	4.931	4.057	4.206
13	6.286	6.390	5.374	5.463
14	14.691	13.562	12.809	11.836

[a]Since REBUS was run with one step, no MOC instantaneous conversion ratios were available and EOC values were not requested, so the REBUS values are averaged over the cycle and the SYNBURN values are instantaneous MOC values.

TABLE VI. Atom % Burnup by Region - Discharge Fraction

Region	REBUS	SYNBURN
1	13.35	13.52
2	13.19	13.35
3	12.87	12.98
4	11.61	11.49
5	9.96	9.60
6	10.58	10.39

TABLE VII. Comparison of ^{238}U, ^{239}Pu, and Fission Product Atom Densities ($\times 10^{24}$)

	BOC					
	REBUS			SYNBURN		
Region	^{238}U	^{239}Pu	F.P.	^{238}U	^{239}Pu	F.P.
1	0.004843	0.0008050	0.0003029	0.004829	0.0008115	0.0003046
2	0.005654	0.0009400	0.0003488	0.005637	0.0009474	0.0003507
3	0.003773	0.0006280	0.0002263	0.003763	0.0006330	0.0002270
4	0.005693	0.0009484	0.0003041	0.005681	0.0009575	0.0003000
5	0.003822	0.0006379	0.0001724	0.003817	0.0006452	0.0001658
6	0.005150	0.001339	0.0002792	0.005127	0.001356	0.0002741
7	0.01331	0.0002834	0.00005539	0.01331	0.0002823	0.00005537
8	0.006042	0.0002159	0.00005969	0.006050	0.0002115	0.00005713
9	0.007057	0.0002473	0.00006697	0.007081	0.0002331	0.00005973
10	0.004790	0.0001627	0.00004312	0.004684	0.0001767	0.00005125
11	0.007109	0.0002156	0.00005096	0.007136	0.0001981	0.00004322
12	0.004765	0.0001269	0.00002694	0.004774	0.0001213	0.00002444
13	0.007216	0.0001418	0.00002389	0.007221	0.0001382	0.00002333
14	0.01354	0.00009977	0.000007706	0.01353	0.0001077	0.00001050

	EOC					
	REBUS			SYNBURN		
Region	^{238}U	^{239}Pu	F.P.	^{238}U	^{239}Pu	F.P.
1	0.004608	0.0007438	0.0005979	0.004600	0.0007491	0.0005929
2	0.005381	0.0008690	0.0006901	0.005375	0.0008753	0.0006826
3	0.003596	0.0005821	0.0004463	0.003592	0.0005861	0.0004420
4	0.005469	0.0008871	0.0005903	0.005459	0.0008937	0.0005857
5	0.003707	0.0006037	0.0003246	0.003697	0.0006084	0.0003249
6	0.005020	0.001222	0.0005330	0.004996	0.001234	0.0005333
7	0.01317	0.0003836	0.00008237	0.01317	0.0003854	0.00008362
8	0.005928	0.0002900	0.00009153	0.005942	0.0002822	0.00008702
9	0.006928	0.0003323	0.0001026	0.006964	0.0003118	0.00009083
10	0.004625	0.0002183	0.00006591	0.004592	0.0002348	0.00007808
11	0.007004	0.0002892	0.00007694	0.007038	0.0002687	0.00006631
12	0.004706	0.0001697	0.00004015	0.004711	0.0001669	0.00003807
13	0.007153	0.0001915	0.00003545	0.007154	0.0001905	0.00003566
14	0.01350	0.0001368	0.00001119	0.01348	0.0001494	0.00001550

TABLE VIII. Comparison of Regional Breeding Ratio Contributions

	REBUS	SYNBURN
Inner Core	0.35296	0.34826
Outer Core	0.17583	0.17043
Axial Blanket	0.19871	0.19595
Radial Blanket	0.36127	0.37220
TOTAL	1.08877	1.08684

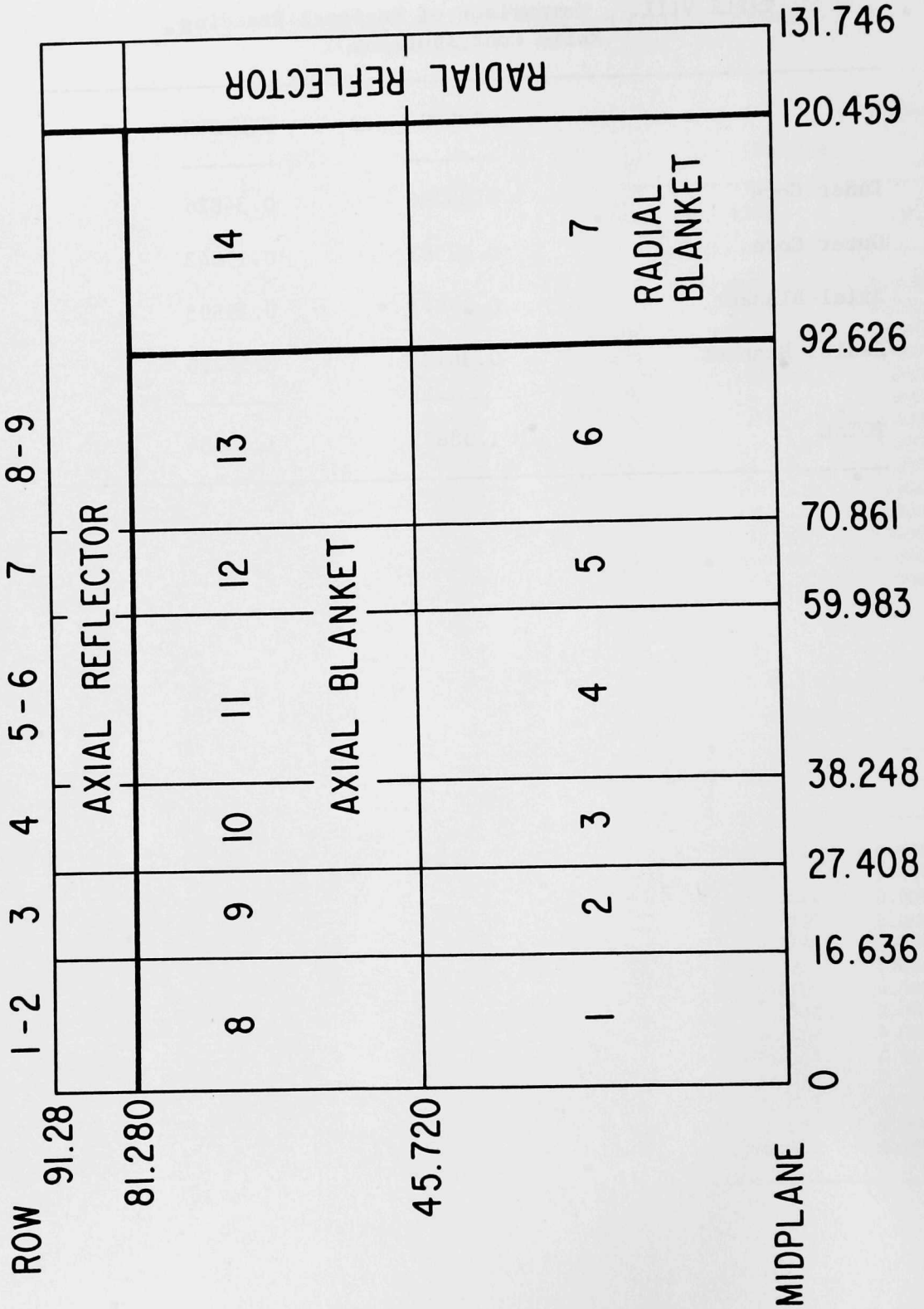

Fig. 3. Simplified R-Z Reactor Model for SYNBURN and REBUS.

Table IX summarizes results from a second REBUS run where region-dependent cross-sections were used, the core was divided axially and many more nuclear reactions were taken into account than could be handled by SYNBURN and the treatment of different individual and grouped fission products. Fig. 4 depicts the geometry of this case. Since the region configuration was different, individual regions could not be compared and so only core and blanket zones appear. The run time was 58 minutes. In this REBUS run, ^{241}Pu β-decay was included whereas in the SYNBURN run and the previous REBUS run, it was not. When a new SYNBURN case was done with ^{241}Pu β-decay included, the breeding ratio from SYNBURN dropped to 1.06592 with regional contributions as follows: inner core 0.3385; outer core 0.1586, axial blanket 0.1960; and radial blanket 0.3728. The comparison is reasonably good for core and blanket average values but the agreement is not so good for details especially in the blankets. However, the comparison provides some justification for using SYNBURN and other synthesis codes for computations, such as parametric studies, where great accuracy may not be required.

VII. DESCRIPTION OF OUTPUT

Values for the following are printed out at each time point for all radial problems and slab calculations.

1. K-effective

2. Normalized region-averaged flux for each group

3. Normalized region-averaged one-group flux

4. Active isotope atom densities for each material-stage

5. Region averaged atom densities, including inactive isotopes and control

6. Median energies in core regions within the calculational channel for source, absorption and flux

7. Regional power increments for all time points (printed only at the first time step in the radial calculation)

8. Flux integrals for each region in each group j in the direction of calculation

   in slab geometry: $\int \phi_j(z)dz$

   in cylindrical geometry: $2\pi \int r\phi_j(r)dr$

   in spherical geometry: $4\pi \int r^2\phi_j(r)dr$

9. Effective buckling in direction of calculation for each region and group calculated from

TABLE IX. REBUS Results With Altered Geometry, etc.

	^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
	Core Charge Masses					
Inner Core	6.6	953.4	168.3	48.0	25.3	6.0
Outer Core	4.9	706.4	203.3	58.0	30.5	7.2
	Reactor Loading B.O.C.					
Inner Core	15.1	2748.4	469.9	158.2	60.0	19.7
Outer Core	12.2	2064.3	554.8	183.3	76.6	23.1
Axial Blanket	8.6	4948.9	120.2	7.5	0.2	--
Radial Blanket	30.8	16055.8	236.3	8.3	0.1	--
	Reactor Loading E.O.C.					
Inner Core	11.2	2639.8	438.4	170.4	47.6	21.0
Outer Core	10.1	2010.4	505.0	191.0	64.1	24.2
Axial Blanket	7.7	4887.3	161.9	11.9	0.5	--
Radial Blanket	29.1	15943.2	322.9	13.3	0.3	--

Power Fractions

	B.O.C	E.O.C.
Inner Core	0.50926	0.47237
Outer Core	0.37416	0.37409
Axial Blanket	0.04869	0.06419
Radial Blanket	0.06790	0.08936

	Regional Breeding Ratio Contribution
Inner Core	0.33740
Outer Core	0.16834
Axial Blanket	0.19164
Radial Blanket	0.34835
TOTAL	1.04572

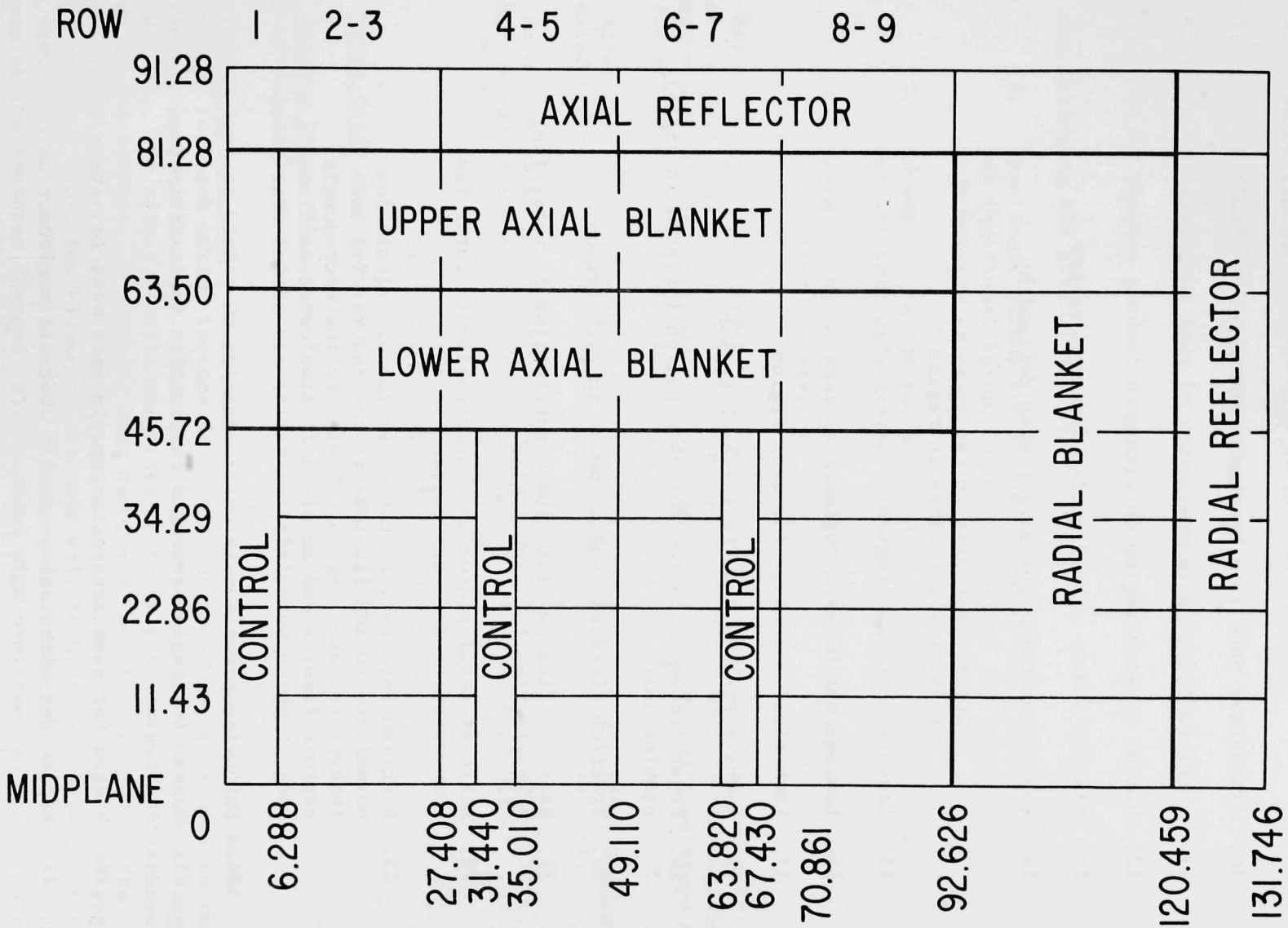

Fig. 4. R-Z Reactor Model for REBUS With Altered Geometry

$$B_j^2 = \frac{\int \nabla^2 \phi_j(\xi)\, d\xi}{\int \phi_j(\xi)\, d\xi}$$ (where ξ is the spatial index)

10. Transverse bucklings by region and group

11. Outer radius of each region

12. Volume of each region in liters

13. Fission source / k-effective for each region

14. Conversion ratio for each region defined by

$$\frac{\text{fertile captures in region}}{\text{fissile absorptions in region}}$$

15. Leakage into each region

16. Leakage out of each region

17. Peak fission density for each region

18. Average fission density for each region

19. Power increment factor for each region (printed in radial problem only)

20. Fraction of total reactor power in each region

21. Ratio of fissile plutonium concentration to total fuel concentration for each region

22. Ratio of total plutonium concentration to total fuel concentration for each region

23. Reaction integrals in the direction of calculation normalized to 100 fissions in the reactor for each isotope and each region. These are the macroscopic capture fission and nu sigma fission cross-sections, each of which is multiplied by (8) and summed over groups

24. Pointwise power densities in watts/cc

25. Masses for region-averaged composition for each isotope and region

26. Masses for each active isotope in each material-stage

27. Masses for each isotope summed over all regions

28. Sum of fissile mass in non-core regions

29. Sum of heavy atom mass in non-core region

30. Sum of fission product mass in non-core regions

31. Sum of fissile mass in core regions

32. Sum of heavy atom mass in core regions

33. Sum of fission product mass in core regions

The following are given for each time step:

1. Fractional burnup and burnup in MWD/MT over the step and fractional burnup from the start of the cycle for each material-stage

2. Conversion ratios for each material-stage, region, and for the core as averages over the time step

3. Change in total mass of specified isotopes summed in each region over the step

After the last time step of the last slab or radial problem, breeding ratios for each time step and for the cycle are given as well as the regional numerator and denominator components and the components for the time steps and for the entire cycle for each region.

After the last time point of the last calculation, core, radial blanket and axial blanket charge and discharge fraction masses are printed out.

In a problem utilizing the startup option, the output for each cycle during startup is the same as the output for an equilibrium problem.

VIII. INPUT DATA SPECIFICATIONS

A. Title Card: One Card; 80 columns of BCD data.

B. Isotope List: This card lists up to 20 isotopes whose cross sections are to be read into core storage. The format is 20I3.

C. Active Isotope List: On this card, the user specifies two pieces of information for each active isotope (and only for those isotopes used in the program) in a specific order. The first number is an integer N in the range: $-3 \leq N \leq +3$. The sign indicates what role is to be played in the enrichment search by the isotope. A negative sign denotes a fuel isotope (e.g., the plutonium isotopes). A plus sign denotes a diluent isotope (e.g., ^{238}U and ^{235}U). Each of these two blocks of isotopes will be held in constant ratio when varied in an enrichment search. In the code, enrichment is defined as the sum of the isotopes flagged with a minus sign over the sum of the isotopes

flagged with either a plus or a minus sign. A plus or minus two indicates a fissile isotope in the breeding ratio computation, and a plus or minus one indicates a fertile isotope. A zero is used for fission products and the burnable poison. A plus or minus three indicates that the isotope is to play no role in the breeding ratio calculation. The second number is the cross section set isotope number for the particular active isotope. The burn matrix handles twelve isotopes in the following order: ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, fission product pairs, burnable poison. The user must specify the above two pieces of data for each of the above active isotopes whose cross sections have been read into core and only these; and they must be specified in the above order as follows: use one I3 field for each of the two pieces of data (the card format is 24I3) for each active isotope, and leave blank the spaces corresponding to the unused isotopes in between. It should be clear that the card has twelve sets of two I3 fields apiece and that each of these sets corresponds, in order, to each of the twelve isotopes above. In the first I3 field, the user specifies the integer flag. In the second, the number of the active isotope is specified.

D. Set of Fixed Point Data: The user specifies individual elements in a singly indexed array. He specifies from one to ten consecutive data elements in the array on each card, and he uses as many cards as necessary. The format of each card is: (I2, I2, I8, 10I6). The number of consecutive array elements to be given on the card goes in the first I2 field; this is an integer from one to ten. In the second I2 field, the user sets zero or blank if the card is not the last card in the fixed point data set, and one if it is. In the I8 field is specified the address in the array of the first of the array elements on the card. The address for each data element is given below. The 10 I6 fields are for the array elements. The fixed point data array is entirely zeroed at the beginning of the program, and therefore any element which is to be set to zero may be ignored. An explanation of each address in the array (call it L) follows:

L1: Number of time steps.

L2: Number of materials in a radial problem for the synthesis solution, or the number of materials for the geometry specified in a 1-D solution. If it is not set, L2 is set equal to L3.

L3: Exactly the same as L2, but for the number of regions. This must be set.

L4: Number of regions in axial geometry for a synthesis solution.

L5: Unused.

L6: Number of regions in which control search is carried out, if any.

L7: Set to one for a synthesis problem, to zero for a 1-D problem.

L8: Inner boundary condition for radial geometry in a synthesis problem or for the geometry specified in a 1-D problem. Option 2 applies only to a 1-D run.

0: grad $\phi_j = 0$

1: $\phi_j = 0$

2: $\phi_j + \omega_j'$ grad $\phi_j = 0$ (enter ω_j values later)

4: same as 2, only ω_j' is calculated by the code

L9: Same as above for the outer boundary.

0: $\phi_j = 0$

1: grad $\phi_j = 0$

2: $\phi_j + \omega_j'$ grad $\phi_j = 0$ (enter ω_j' values later)

4: same as 2, only ω_j' is calculated by the code

L10: Inner boundary condition for all the slab problems in a synthesis problem; the options are the same as in L8.

L11: Outer boundary condition for all the slab problems in a synthesis problem; the options are the same as in L9.

L12: Geometry specifications for a one-dimensional problem: 1 for a slab; 2 for a cylinder; 3 for a sphere.

L13: Limit on the number of flux solution iterations; if not set, it will be sixty.

L14: Limit on the number of enrichment and control search iterations; if not set, it will be fifteen.

L15: Set to zero for the analytic solution of the burn equations and to one for the series solution where the analytic is used for the remainder of the region only and to two for the series solution where the analytic is used for the remainder of the problem and to three if the series solution is to continue to convergence.

L16: Set to zero if there is no material shuffling (none is allowed in a synthesis problem in radial regions where a slab calculation is to be done; it is up to the user to be certain his input specifications are consistent) and set to one if there is.

L17: Set to zero if no power peak matching is to be done and set non-zero if there is.

L18: Set to zero if the input cycle time is to be adjusted; set to one if the input time is not to be adjusted.

L19: Set to one if an unpoisoned k-effective at beginning-of-life is to be searched for such that there is a k-effective of one at the end of the cycle with no control in the system.

L20-39: L3 numbers giving the number of mesh intervals in each radial region in a synthesis problem or in the geometry specified in a 1-D problem. Start with the inner region for a cylinder and with the bottom on a slab.

L40-59: L4 numbers giving the number of intervals in each axial region for a synthesis problem; start with the bottommost region.

L60-79: L3 numbers indicating the role that each region is to play in an enrichment search; for the radial regions in a synthesis problem or for the geometry specified in a 1-D problem. Set the number for the region non-zero if it is a core region with active isotopes (e.g., as opposed to a control zone). Set to one if the region is the first zone in an enrichment zone and set to two if not. The latter requirement is for the peak power matching option.

L80-99: L4 numbers describing each slab region in a synthesis problem (this applies to all the slab calculations). Start with the bottommost region. Set to one if the region is the core zone or, in a radial blanket calculation, if it is that zone adjacent to the core which is included in the radial calculation. Set to two if this region is to receive the maximum amount of control which the region described previously contained during the radial calculation (i.e., the upper axial blanket zones). Set to three if the bucklings for each group in this region are to be set to zero (this option is available only for the reflector zones, and it excludes the previous option). Set to zero if none of the above conditions applies to the region.

L100: Maximum number of downscatter groups in cross section set.

L101: Set non-zero if transverse bucklings are input for the first radial problem in a synthesis problem or for the geometry specified in a 1-D problem. These bucklings are for each region, group, and time point (i.e., if there are N time points, there are N - 1 time steps). These values will be read in after the material data below.

L102: In a synthesis problem, set this non-zero if the code is to punch axial bucklings for each time point, group, and radial region. This will be done for each set of slab calculations, so the user must select the ones for the last set.

L103: For the first radial problem in a synthesis run or in the geometry specified in a 1-D run, set non-zero if transverse bucklings for each time point and region, but not each group, are to be read in. These will begin with A60.

L104: Set non-zero if group-dependent but time- and region-independent transverse bucklings are read in starting at A60. This is for the first radial problem in a synthesis run or for the geometry specified in a 1-D run.

L105: Set non-zero if region-dependent but time- and group-independent transverse heights or radii are read in starting at A60. This is for the first radial problem in a synthesis run or for the geometry specified in a 1-D run.

L106: Set equal to the maximum number of times a set of slab calculations is to be done after the radial problem; if it is to be done after all radial calculations, merely set to some large number, such as ten or fifteen.

L107: Set non-zero if transverse bucklings are to be read in that are group- and region-dependent but not time-dependent. This is for the first radial problem in a synthesis run or for the geometry specified in a 1-D run. These are entered at A60.

L108: Set non-zero if power increment factors are input for each region and time point. This is for the first radial problem in a synthesis run. These will begin at A580.

L109: Set non-zero if group- and time-independent transverse bucklings are specified for the slab problems.

L110: Set non-zero if the user wants atom densities and fluxes printed out for each set of radial and axial problems run; these will be printed out for each time point. If left at zero, only the complete edits for the last radial and axial problems will be printed.

L111: Set non-zero if the startup option is to be used. Set to one if the full amount of control from the equilibrium cycle calculation is to be used. Set to two if the control at each time point in the cycle is to be altered as described in A14.

L112: Set to zero if the radial blanket is to be loaded with enough ^{235}U to produce the same average power as at the beginning of life equilibrium. This average amount of ^{235}U will be loaded into all stages for the initial loading. Set to N if the radial blanket is not to be reloaded until the Nth cycle.

L113: This is the number of cycles during which the equilibrium control is multiplied by exactly A14 at each time point. After the L113th cycle, the factor is changed as is described in A14. If the A14 factor is not used, set to zero.

L120-139: Enter L3 numbers here for a snythesis problem. If the user wants a slab calculation for a particular radial region, set the data element corresponding to this region to zero. If a slab problem is not to be done (as for a reflector), set to one. Begin with the innermost radial region.

L140-159: Enter the numbers of the radial materials consecutively in ascending order here over which the discharge burnup is to be averaged.

L160-171: Twelve numbers, for each of the active isotopes (in the same order as on the active isotope list), are entered here. Set non-zero if it is desired to include this isotope in a group (e.g., the plutonium isotopes) such that the total mass of this group is calculated for each time step and for each region.

E. Set of Floating Point Data: Again the user specifies elements in a singly indexed array. He can specify from one to five consecutive elements in the array on each card, and he uses as many cards as necessary. The format of each card is: (I2, I2, I8, 5E12.5). The fields perform the same function as with the fixed point data. The floating point array (call it A) is again entirely zeroed before any data are read in. An explanation for each element of the array follows:

A1: Convergence limit for the flux solution; i.e., the criterion for convergence is that $(k^{\ell} - k^{\ell-1})/k^{\ell}| <$A1 where k^{ℓ} is k-effective on the ℓth iteration.

A2: Convergence limit on enrichment and control searches; i.e., condition for convergence is that $|k_{eff} - 1| <$ A2.

A3: Convergence limit on the fractional discharge burnup averaged over the specified materials. The condition for convergence is that the absolute value of the difference between the required fractional burnup and the burnup obtained is less than A3.

A4: Convergence limit on the final control in the system. If the final control atom density is less than twice A4 times the initial control guess but greater than zero, then the convergence is satisfied.

A5: The average fractional burnup required in the materials specified.

A6: Initial guess for the equilibrium cycle time or the specified cycle time when no adjustment is to be made.

A7: Total reactor power in watts.

A8: Full core height for a two-geometry run or a 1-D radial run. This must also be the radial blanket height.

A9: Convergence criterion on k-effective in initialization procedure for slab calculations at beginning-of-cycle. As soon as the difference between the k-effectives in any two consecutive calculations during the iterative process is less than this number, the iteration ceases. This criterion applies to the slab problems in the startup option as well.

A10: Source extrapolation factor; if not set, it will be 0.2.

A11: Physical radius of core in a one-dimensional slab problem.

A12: The convergence criterion for the series solution of the burn equations.

A13: Inner radius of the reactor in a 1-D cylinder run or for the cylinder in a two-geometry run or the lower physical dimension of a 1-D slab problem.

A14: If L111 is 2, the equilibrium control will vary at each time point as follows: for the first L113 cycles, the control fraction will be multiplied by A14; for the L113 + Nth cycle the control will be

multiplied by (A14 - 1.) × (M - N)/M + 1, where M is the number of stages in the core materials; this is until the L113 + Mth cycle and from here on the equilibrium control is used.

A15: ^{241}Pu decay constant.

A20-39: The outer radii of each slab region (all slab problems must share these radii), starting with the bottommost, for a two-geometry problem.

A40-59: The outer radii of each radial region (starting with the innermost) for a 1-D cylinder problem or for a two-geometry run, the radii for the radial problem. In a 1-D sphere or slab problem, these are the outer radii for each region starting with the innermost or bottommost.

A60-579: If L103 ≠ 0, enter a transverse buckling value for each region and time point as follows: enter the value for region I and time point J in location A(L), where L = (I - 1) · N + J + 59 and N = number of time steps plus one.

If L104 ≠ 0, enter a buckling value for each group starting with A60.

If L105 ≠ 0, enter a height or radius for each region starting with A60.

If L107 ≠ 0, enter buckling values for each region and group as follows: enter the value for region I and group J in location A(L), where L = (J -1) · 20 + I + 59.

If none of the above indicators are set, enter one height or radius in A60.

A580-719: If L108 ≠ 0, then enter values for the power increment factors for each region and time point as follows: enter the value for region I and time point J in A(L), where L = (J -1) · 20 + 579 + I.

If only region-dependent values are entered, place them in the first M locations beginning with the innermost region at A580, where M is the number of regions.

If one value is used for all regions and time points, enter it at A580.

If no value is entered, all values will be one.

A720: In a two-geometry problem, this is the lower radius of the slabs. It is initially set to zero in the code.

A820-845: Outer boundary extrapolation lengths if L9 = 2.

A846-871: Inner boundary extrapolation lengths if L8 = 2.

A872-891: Buckling values as described in I.

A892-911: Atomic mass for each isotope in ascending order of isotope number.

F. Specification of control distributions: L6 sets of data will be specified here (where L6 $\geq$ 0). Each set starts with a card on which the user specifies the region number (in radial geometry, of course, because this is where the search is done) in which the control for this set resides, and the number of isotopes of both poison and diluent whose atom densities will be specified on succeeding cards. The format of the first card is: 2I6. Place the region number in the first field and the number of isotopes in the second. Next use as many cards as are necessary to specify poison and diluent atom densities in the specified region. The card format is: 3(I6, 6X, E12.5). Therefore, the user specifies a maximum of three atom densities per card. For each density, specify the isotope's number in the I6 field and its density in the E field. The control density input is merely an initial guess for the first control search and therefore the absolute amount of the control is less important although it should be within bounds. But the relative amounts of poison and diluent and the ratios of the control among the regions are the important things specified in these densities. If the isotope is a poison, then use the normal cross-section tape number, and if it is a diluent, use the negative of the tape number for the isotope. The code varies the control atom densities on the assumption that the atom density of the poison corresponds to the same volume fraction as the density of the diluent. During the control search, the ratios of the control densities in one region to that in another are held constant as specified in the input numbers.

G. Material atom densities: The user next specifies the active and inactive atom densities for the first stage of each material. First one card (Format I6) with the total number of materials to be read in. For each material specified, a set of data in the following form is necessary. The first card in the set has the format (3X, I3, I6, E12.6, 6A8). In the first I3 field, specify the number of stages in the material (maximum of ten). In the I6 field, specify the number of isotopes to be read in on succeeding cards. If L16 is zero, enter nothing in the E field; if L16 is one, enter the subassembly volume for the material stages (this is not a single subassembly volume, but the total volume that is taken by all the subassemblies of one stage of this material). Next are 48 columns of BCD material identifier information. If L16 is one, the second card in the set consists of the region number of residence for each stage of the material. The card format is: 10I3. For each stage, in ascending order starting with the first, specify the region in which this stage resides. If L16 is zero, omit this card. Next, use as many cards as are necessary to specify the atom densities for the first stage of this material. Place a maximum of three on each card with the same format as for the control densities above. For each isotope, of course, the cross-section set number and the density will be specified as it was for the poison. The above specifies the complete set of data for each material. The materials here will be numbered in ascending order as they are read in. The user specifies a ratio of enrichments among materials in the first stage, and this is held constant in the enrichment searches unless the peak power matching option is selected.

H. Material-region identification: Input an array on L4 cards, each card with L2 numbers on it (format 20I3). In the first card give the materials in the bottommost axial tier starting with the inner radial region and

proceeding outwards. On the next card give the materials in the second to lowest axial tier going radially outward and so on up to L4 cards, L4 being the number of axial regions.

I. Region-, time-, and group-dependent axial or transverse bucklings: If L101 is one, enter these values in the following manner. The user provides buckling values for each group for the first region and the first time point, then J values (J = number of groups) for the first region and the second time point and so on for all the time points. Then begin with the second region and the first time point and proceed in this manner for all regions and time points. For each region and time, J bucklings are specified with a maximum of six to a card (format: 6E12.5) and when the last group has been specified, skip to the next card. For example, if there are four regions and three time points and twenty-two groups, there will be 48 cards, and not 44. With each set of group-dependent bucklings, the first 18 will go on the first three cards and then the next four on a fourth card. This option will probably only be used when one is using the bucklings punched from a previous problem.

J. Group- and time-independent bucklings for slab problems: The user may specify that group- and time-independent radial bucklings be used for a region in a particular slab problem. If this option is to be used, set L109 $\neq$ 0. Next enter the buckling value for a particular slab corresponding to radial region R in A(871 + R). There can be only one radial buckling for a particular slab problem, although for the different slabs, different bucklings are allowed. The user then has three options to specify the radial bucklings for a given slab. He may elect to use the bucklings calculated in the core and radial blanket for any of the regions in the slab problems. For the axial reflector, he may set the radial bucklings to zero. And if he wishes to specify a buckling that is group- and time-independent but non-zero, he may use this option. Besides the specifications in the fixed and floating point data as described above, if L109 $\neq$ 0, the user here specifies in which regions he wants to use this option. One card is required for each of the L4 slab regions starting at the bottom. On each card (format: 20I3), specify L3 numbers corresponding to each of the radial regions. Set the number to one if the buckling value for this radial region (as specified above) is to be used and to zero if the core bucklings or zero bucklings are used.

K. Breeding ratio area definition: Read L4 cards, each card with L2 numbers on it (format 20I3). Use this L4 by L2 array, which represents all regions to define "areas" over which conversion ratios are averaged. Enter a "1" if it is an inner core zone, "2" for outer core, "4" for axial blanket and "5" for radial blanket. Enter "0" for all other regions (e.g., reflector regions and control zones).

L. The startup option: This input follows the set of normal input for an equilibrium calculation. The user must input a complete set of enrichments and isotopic distributions for each cycle of startup. However, when the input to two or more cycles is identical, then only one data set is required and it is sufficient to indicate that this set is repeated.

First Card: (format 21I3) in first I-field, the number of sets of data; in the next N fields are the data set numbers for each of the N cycles of startup. For example, assume that there are 3 different data sets and that the third set is used in the Mth cycle; then the Mth number in the array will be three.

Second Card: (format 12I3) flags (-1, 0, or 1), the same as in the third card of the equilibrium calculation data to indicate the type of fuel. Enrichment will be defined as the sum of the "minus" isotopes over the total.

For each data set:

1. Radial Problem

   a. Enrichments for each radial region (means of inputting starting guesses for enrichments as well as for ratios of enrichments between zones) format - 6E12.5; use as many cards as necessary.

   b. Isotopic distributions for fuels and non-fuels; 2 cards per region - On first card the distribution for up to 4 fuel (flagged with a minus one) isotopes; on the second the distribution for up to 4 non-fuel isotopes. Format for both is 4E12.5.

2. Slab Problems: 1 set as follows for each slab problem.

   a. Enrichments for each axial region; same format as in 1.a.

   b. Isotopic distribution for fuel and non-fuel isotopes specified for each region as in 2.a.

Last Card: flags defining a different fuel (if any) and if it is the same then a repetition of card 2.

Sample Problem for SYNBURN Equilibrium Calculation:

The following set of data is for a modular reactor. The burn is done in two time steps. There are four radial regions in the synthesis problem and, since the slab problems are done symmetrically, there are three slab regions. There are two regions in which control resides. There will be control in the axial blanket, and there is no transverse leakage in the axial blanket. There is no slab problem done, of course, in the reflector zone. The power peak matching routine is used. There are five downscatters in the cross section set. There are two sets of slab calculations done besides the edit. The burnup is to be averaged over the two core materials. There is a single extrapolated height for the first radial problem. Isotopes 5 and 1 are the poison and isotope 6 is a diluent.

COMPUTER INPUT DATA FORM

COST CODE _____

PROGRAM | PROBLEM | ORIGINATOR | DATE | PAGE | OF

		SAMPLE	PROBLEM																
1	6	5 8	7 21	11 12	13 14	16 17	18 19	20											
+1	17	-2 18	-1 19	-2 13	-1 14						+2 16			21					
4		1	2	4	4	3													
2		6	2	1															
1		17	1																
4		20	15	10	20	4													
3		40	15	15	10														
4		60	1	1	0	0													
3		80	1	2	3														
1		123	1																
1		106	2																
2	1	140	1	2															
5		1	.000001		.001		.001				.02			.1054					
4		6	292.			2.6D9		93.8			.001								
1		5	1.665D-9																
3		20	46.89		85.24		125.24												
4		40	48.89		66.77		124.84				138.29								
1		60	136.23																
3		580	1.07		1.07		1.2												

COMPUTER INPUT DATA FORM

COST CODE _____

PROGRAM | PROBLEM | ORIGINATOR | DATE | PAGE | OF

5	89.2	12.	10. 2	23.	16.	0.	
5	89.7	0.	0.	241.	242.	235.	
5 1	90.2	238.	239.	240.	55.	236.	
1	3						
5		.001	-6	.0.0.0.2.6	1	.0.0.0.2.5	
2	3						
5		.0.0.1.1	-6	.0.0.0.2.8.6	1	.0.0.0.2.7.5	
6							
3	9		INNER CORE MATERIAL				
6		.0.1.2.9.8.3.8	1	.0.0.8.9.0.7.6	20	.0.0.9.3.8.1.2	
14		.0.0.0.0.5.3.8	13	.0.0.0.1.7.5	19	.0.0.0.2.9.6.2	
18		.0.0.0.8.2.1.3	17	.0.0.7.5.3.8.6	16	.0.0.0.0.2.2.7	
3	9		OUTER CORE MATERIAL				
16		.0.0.0.0.2.0.6	17	.0.0.6.8.6.5.6	18	.0.0.1.2.3.1.9	
19		.0.0.0.4.4.4.3	13	.0.0.0.2.6.2.5	14	.0.0.0.0.8.0.8	
1		.0.0.8.9.0.5.7	6	.0.1.2.9.8.3.8	20	.0.0.9.3.8.1.2	
3	5		RADIAL BLANKET				
16		.0.0.0.0.3.3.2	17	.0.1.1.0.4.2.5	7	.0.2.2.1.5.1.4	
6		.0.0.9.4.9.1.2	20	.0.0.6.8			
1	4		RADIAL REFLECTOR				

COMPUTER INPUT DATA FORM

COST CODE _____

PROGRAM | PROBLEM | ORIGINATOR | DATE | PAGE | OF

6	.004382	8	.007124	12	.048805
11	.013005				
3 5		AXIAL BLANKET			
16	.000267	17	.008846	1	.008913
6	.012938	20	.009312		
1 2		AXIAL REFLECTOR			
6	.015342	20	.025401		
1 2 3 4					
5 5 3 4					
6 6 6 4					
1 2 5 0					
4 4 5 0					
0 0 0 0					

Sample Problem for Startup Option:

There are 2 sets of data, the first of which applies to the first cycle, and the second of which applies to all subsequent cycles. The fuel is plutonium. The enrichments in the inner and outer core zones are 17 and 27 per cent respectively. There are 5 radial zones and 3 axial zones. The plutonium distribution in the core is 60:24:12:4 for the first cycle and 66:27:5:2 in the second. No ^{235}U is considered in the depleted uranium in the core in the R.B. loading in the second cycle. The inner R.B. is 1.6% ^{235}U and the outer R.B. is 1.9% in the first cycle.

COMPUTER INPUT DATA FORM

COST CODE _____

PROGRAM | PROBLEM | ORIGINATOR | DATE | PAGE | OF

1	2	3	4	5	6	7	8
12	12	2	2	2	2	2	2
+1	-2	-1	-2	-1		+2	
	.17		.27				
	.6		.24		.12	.04	
	1.						
	.6		.24		.12	.04	
	1.						
	.984		.016				
	.981		.019				
	.17						
	6		.24		.12	.04	
	1.						
	1.						
	1.						

COMPUTER INPUT DATA FORM

COST CODE __________

PROGRAM | PROBLEM | ORIGINATOR | DATE | PAGE | OF

.27			
.6	.24	.12	.04
1.			
1.			
1.			
.984	.016		
.984	.016		
.984	.016		
.981	.019		

COMPUTER INPUT DATA FORM

COST CODE ____________

PROGRAM PROBLEM ORIGINATOR DATE PAGE OF

Row	Cols 1–10	Cols 11–20	Cols 21–30	Cols 31–40	Cols 41–50	Cols 51–60	Cols 61–70	Cols 71–80
1								
2	.17	.27						
3	.66	.27		.05	.02			
4	1.							
5	.66	.27		.05	.02			
6	1.							
7								
8	1.							
9								
10	1.							
11								
12								
13	.17							
14	.66	.27		.05	.02			
15	1.							
16								
17	1.							
18								
19	1.							
20	.27							

COMPUTER INPUT DATA FORM

COST CODE _____

PROGRAM	PROBLEM	ORIGINATOR	DATE	PAGE	OF

1	2	3	4	5	6	7	8
1 2 3 4 5 6 7 8 9 0	1 2 3 4 5 6 7 8 9 0	1 2 3 4 5 6 7 8 9 0	1 2 3 4 5 6 7 8 9 0	1 2 3 4 5 6 7 8 9 0	1 2 3 4 5 6 7 8 9 0	1 2 3 4 5 6 7 8 9 0	1 2 3 4 5 6 7 8 9 0
.66	.27	.05	.02				
1.							
1.							
1.							
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COMPUTER INPUT DATA FORM

COST CODE ______________________

PROGRAM PROBLEM ORIGINATOR DATE PAGE OF

	1	2	3	4	5	6	7	8
1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0

+1 -2 -1 -2 -1 +2

	1	2	3	4	5	6	7	8
1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0 1 2 3 4 5 6 7 8 9	0

SYNBURN DICTIONARY

A(1)	ESP1	Epsilon on flux solution.
A(2)	ESP2	Epsilon for control and enrichment searches.
A(3)	BUEPS	Epsilon for burnup.
A(4)	PEPS	Epsilon on final control in system.
A(5)	BULIM	Avg. discharge burnup required.
A(6)	TBURN	Current burn cycle time.
A(7)	POWR	Total reactor power.
A(8)	HITE	Core height.
A(9)	ESP3	Convergence criterion for slab problems.
A(10)	THETA	Source extrapolation in FLUX.
A(11)	RTRANS	Physical radius for slab.
A(12)	BEPS	Epsilon for series solution of burn equations.
A(13)	RA1	Lower radius of a slab.
A(14)	SPSN	Control multiplier for startup option.
A(15)	A15	^{241}Pu decay constant.
A(16)	A16	Unused.
A(17)	A17	Unused.
A(18)	A18	Unused.
A(19)	A19	Unused.
A(20)→A(39)	ROUT(20)	Outer radius of each region.
A(40)→A(59)	ROUTA(20)	Outer radius of each region in opposite geometry for a synthesis problem.
A(60)→(579)	BUCK2(20,26)	Region- and group- dependent transverse buckling.
A(580)→A(719)	PFAC(20,7)	Power increment factors for each region and time.
A(720)→A(819)	R(100), RR1	R is radius at each point; RR1 is first radius.

A(820)→A(845)	OMEG1(26)	Outer boundary extrapolation lengths.
A(846)→A(871)	OMEGA(26)	Inner boundary extrapolation lengths.
A(872)→A(891)	B2R(20)	Transverse bucklings for slabs.
A(892)→A(911)	AMASS(20)	Atomic mass of isotopes.
DENS(20,10,20)		Atom densities for active and inactive isotopes for each material-stage. (material x stage x nuclide) where nuclides are in ascending order.
ADENS(20,10,12)		Atom densities for active isotopes for each material stage. (material x stage x nuclide) where active nuclides are in burn matrix ordering.
DIF(20,26)		Macroscopic diffusion coefficient. (region x group).
VUSIG(20,26)		Macroscopic nu sigma fission (region x group).
SIGA(20,26)		Macroscopic capture plus fission (region x group).
AC3(20,26)		Scratch. Used in FLUX for finite difference coefficients, and several other places. Used in BUCKL to store B_j^2 in direction of calculation.
AC4(20,26)		Scratch. Used in FLUX for finite difference coefficients and elsewhere.
AC5(20,26)		Same as AC4(20,26).
AC6(20,26)		Same as AC4(20,26).
FLP(150)		Finite difference coefficients.
A1(150)		Scratch.
A2(150)		Scratch.
DELTA(150)		Used in calculating finite difference coefficients.
TRANS(20,15,26)		Macroscopic downscatter array (reg x scatter x group) TRANS (I,K,J) is the scattering to group J from group L+K-1 where L is the group number of the highest energy group which can scatter into J (L is the larger of 1 and NDS).

SVU(26,20)	Microscopic nu sigma fission (group x nuclide).
GNU(26,20)	Number of neutrons per fission (group x nuclide).
SSA(26,20)	Microscopic capture plus fission (group x nuclide).
SST(26,20)	Microscopic sigma transport (group x nuclide).
STR(26,15,20)	Microscopic downscatter array (group x downscatter x nuclide).
B2JRT(20,26,7)	Storage array for radial bucklings for slab problems. (Region x group x time point)
B2JZT(20,26,7)	Storage array for axial bucklings for radial problems. (Region x group x time point)
TCAPT(12,20,6)	Storage array for A matrix elements-capture. (isotope x region x time point)
TFISS(12,20,6)	Storage array for A matrix elements-fission. (isotope x region x time point)
DENA(19,20,20)	Storage for stage one atom densities for slab problems. (axial region x nuclide x radial region)
ADRO(20,10,12)	Atom densities for time zero in startup option for radial zones (region x stage x active isotope).
ADZO(20,10,12)	Atom densities for time zero in startup option for axial zones. (region x stage x active isotope)
BRAN(20,6,20)	Numerator contribution for breeding ratio (axial region x time point x radial region).
BRAD(20,6,20)	Denominator contribution for breeding ratio (axial region x time point x radial region).
CONCM(20,20)	Region averaged atom densities (region x nuclide).
PHI(100,26)	Pointwise fluxes (point x group).
SCAP(26,12)	Microscopic capture cross-section (group x nuclide). Where nuclides are in burn matrix order.
SFIS(26,12)	Microscopic fission cross-section (group x nuclide).

CAPT(12,20)	Collapsed capture cross-section for burn matrix (nuclide x region).
FISS(12,20)	Collapsed fission cross-section for burn matrix (nuclide x region).
URAT(20,4)	Fraction of total of "plus" isotopes in each of the isotopes in the block (material x nuclide).
PURAT(20,4)	Same as above for "minus" isotopes.
BUSUM(20,10)	The sum of the fractional burnup since the start of the cycle for each material stage (material x stage).
FLX(20,26)	Region-averaged flux (region x group).
FNOR(20,7)	Flux normalization factor for radial region x time point.
PSN(20,20)	Poison atom densities (region x nuclide).
DIL(20,20)	Diluent atom densities (region x nuclide).
DENR(20,20)	Storage for stage one atom densities for radial problem (region x isotope).
TFLX1(20,6)	One-troup flux (region x time step).
SUMX(20,10)	Scratch.
BRRN(6,20)	Accumulator for breeding ratio numerator. (time step x region)
BRRD(6,20)	Accumulator for breeding ratio denominator. (time step x region)
FMID(20,26)	Mid point fluxes (region x group) in startup option.
D(150)	Total source into current group in FLUX.
BETA(150)	FLUX calculation coefficients.
A3(150)	Scratch.
CHI(26)	Fission spectrum.
DELR(20)	Mesh spacing by region.
SOP(20)	Guess of fission source for next iteration at inner interface of each region. Derived from SOPP by source extrapolation.

SOUP(100)	Normalized fission source at each space point at the end of an iteration in FLUX.
RI(100)	RI(I) = R(I)**P
SOPP(20)	Fission source at inner interface of region I is stored in SOPP(I-1). This is the pointwise fission source calculated from $\Sigma_g \nu\Sigma_{f_g}\phi_g$ normalized to an integral of 1.0.
SONME(20)	Isotope name from XSISO.
EIG(3)	Three currently used k-effectives in enrichment search.
XPI(3)	Three currently used enrichment factors in enrichment search.
SUBVOL(20)	Total volume for a stage of a material.
FLX1(20)	One group average flux for each region.
TAU(20)	Regional flux-time.
BCD(10)	Problem ident. array.
ELETHR(26)	Upper lethargy limit of each group.
SUMUPU(20)	Total fuel atom density for each material.
ENRICH(20)	Sum of "minus" nuclides in stage one divided by SUMUPU for each material.
FPPS(20)	Fission product pair concentration in first stage of each material.
BPSN(20)	Burnable poison concentration in first stage of each material.
VOLR(20)	Volume of each region.
RTR(20)	In a synthesis problem, this is the "physical radius" for each slab in order to give correct volumes.
PTM(7)	Control fraction as a function of time.
G(26)	Scratch.
AA(26)	Scratch.
RL(26)	Scratch.

COC(5)	Scratch.
COD(5)	Scratch.
ABC(5)	Scratch.
ABD(5)	Scratch.
RBC(5)	Scratch.
RBD(5)	Scratch.
REC(6)	Scratch.
STEN(20)	Scratch.
G1P(20)	Scratch.
DP(20)	Scratch.
P	0. for slabs; 1. for cylinders; 2. for spheres.
EIGEN	Current k-effective of system.
PIM	1. for slabs (2. for reflective slab problems); 2π for cylinders; 4π for spheres.
W	K-1 in CRIT.
C15	Control fraction on previous pass in CRIT.
XP1P	Current control fraction in CRIT.
WS	Used in control search in CRIT.
PDEL	Change in the control fraction; used in PMIXER.
XP1E	Current control fraction.
ANORM	Flux normalization factor.
FNORM	Core average fission density calculation in radial problem for flux normalization in slab.
TIME	The time in seconds over which the system is burned multiplied by 10^{-24}.
BU2	Discharge average burnup.
TBURN1	Previous burn cycle time.

PLAST		Last positive control fraction over cycle.
PTMAX		Maximum control fraction over cycle.
SERCHK		K-effective required if different than one.
DEDK		Rate of change of enrichment with k-effective.
DPDK		Rate of change of control with k-effective.
SOU(100)		Source guess for next iteration in FLUX. Derived from SOUP by source extrapolation.
L(1)	NOSTEP	No. of time steps.
L(2)	MAA	No. of materials in opposite geometry for synthesis.
L(3)	MAXL	No. of regions in opposite geometry for synthesis.
L(4)	M	No. of regions.
L(5)	MA	No. of materials.
L(6)	N8	No. of control regions.
L(7)	N2D	Indicator for synthesis problem if non-zero.
L(8)	NIBA	Inner bound condition for opposite geometry.
L(9)	NOBA	Outer bound. cond. for opposite geom.
L(10)	NIB	Inner bound. cond.
L(11)	NOB	Outer bound. cond.
L(12)	N1	Geometry; = 1 for slab; = 2 for cycle; = 3 for sphere.
L(13)	MICT	Maximum no. of iterations in flux solution.
L(14)	MICT2	Maximum no. of iterations in control search.
L(15)	NOPT	Option for solution of burn equations.
L(16)	NSHF	Option for material shuffling.
L(17)	NPOW	Peak power adjustment option.
L(18)	NTAD	Time adjustment option.

L(19)	KSR	Set for non critical k-effective search.
L(20)→L(39)	NINTA(20)	No. of mesh points for each region for opposite geom.
L(40)→L(59)	NINT(20)	No. of mesh points in each region.
L(60)→L(79)	NCOREA(20)	Region identifiers in opposite geom.
L(80)→L(99)	NCORE(20)	Region identifiers.
L(100)	NDS	Max. no. of downscatters.
L(101)	L75	Switch for region, group, and time dependent transverse B^2.
L(102)	L76	Switch for punching axial bucklings.
L(103)	L77	Switch for region and time, not group dependent transverse B^2.
L(104)	L78	Switch for group and time not region dependent transverse B^2.
L(105)	L79	Switch for region, not time and group dependent transverse B^2.
L(106)	LSLAB	Max. no. of slab-radial iterations before edit.
L(107)	L81	Switch for group and region, not time dependent transverse B^2.
L(108)	L82	Switch for region and time dependent power increment factors.
L(109)	L83	Switch for group and time dependent B_r^2 for slabs.
L(110)	LPR	Print out option.
L(111)	LST	Startup option.
L(112)	L98	Radial blanket fissile loading in startup option.
L(113)	NOCYC	No. cycles for control multiplier in startup option.
L(114)	L114	Unused.
L(115)	L115	Unused.

L(116)	L116	Unused.
L(117)	L17	Unused.
L(118)	L118	Unused.
L(119)	L119	Unused.
L(120)→L(139)	I2D(20)	Set non-zero for every rad. reg. where a slab is done.
L(140)→L(159)	NB(20)	Reg. numbers for burnup average.
L(160)→L(171)	ISOLST(12)	Set non-zero for specified isotopes.
LBR		Switch to tell when to calculate breeding ratio.
MSREG(20,10)		Region of residence for each material stage (material x stage).
MSREG1(20,10)		Region of residence for each material stage (material x stage) for opposite geometry.
NSTGA(20,20)		No. of stages for each material in axial geometry (radial region x axial region).
LB2R(20,20)		Indicator to tell where to use input B_r^2 in slabs.
LB(20)		
NOSTG(20)		No. stages in each material.
ISAC(12)		Fissile-fertile indicator.
ISOACT(12)		Active isotope cross-section tape numbers.
MATN(20)		Cross-section tape isotope numbers.
MR(12)		Index on MATN array for each active isotope.
NSTGR(20)		No. of stages for each radial material.
ID(26)		Scratch.
LAA(26)		Highest group from which scatter occurs.
NISO		No. isotopes.

NOG	No. groups.
N2	NIB+1.
N3	NOB+1.
NM1	N-1.
NM2	N-2.
MM1	M-1.
N	No. of mesh points.
NCRIT	Transfer indicator in CRIT.
ISERCH	Control search status indicator, = 1 in progress, = 2 failed, = 0 converged.
NSERCH	Enrichment search status indicator, = 1 in progress, = 2 failed, = 0 converged.
IT	Switch in burn calculation.
NTM	Current time step.
MAXSTG	Max. number of stages.
NF	Switch in burn matrix calculation.
ISTAR	Switch in startup calculation.
M7	Input unit no.
KINO	Unused.
IPASS	No. of iterations in equilibrium calc. or cycles in startup.
IRA	The radial region index in slab problems.
NSI	Switch.
NEDIT	Switch to turn on edit.
ICT2	Iteration counter on searches.

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